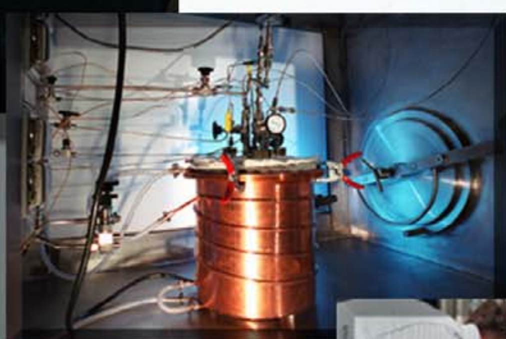
DOE/NE-0123

International Nuclear Energy Research Initiative

2002 Annual Report



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U.S. Department of Energy
Office of Nuclear Energy, Science and Technology
1000 Independence Avenue, S.W.
Washington, D.C. 20585

Contents

Foreword

The International Nuclear Energy Research Initiative (I-NERI) was established by the U.S. Department of Energy (DOE) in fiscal year (FY) 2001 as a mechanism for coordinating international research and development (R&D) on next-generation nuclear energy systems known as *Generation IV*. I-NERI was established in response to recommendations of the Presidents' Committee of Advisors on Science and Technology (PCAST) in the Committee's 1999 report entitled *Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation*. I-NERI along with the Nuclear Energy Research Initiative (NERI) and the Generation IV Nuclear Energy Systems Initiative, are key elements of the Federal effort to foster global cooperation in development of advanced nuclear energy technology.

A key purpose of I-NERI and its predecessor initiative, NERI, is to apply merit-based, competitive methodologies comparable to those in use by other U.S. government science agencies such as the National Science Foundation, for solicitation, evaluation, and selection of projects. Adapting competitive procurement methodologies for solicitation and selection of projects that involve international government funding agencies and international proposal teams comprised of federal laboratories, universities, and industry participants, has proven to be both a significant challenge and a rewarding experience for the participating international funding agencies as well as for participants in the joint R&D proposals and projects.

In less than a year of project performance, the I-NERI program has made substantial progress toward achieving its goals of addressing potential technical and scientific obstacles to the continued global implementation of nuclear energy, establishing of international collaborations for development of advanced nuclear technology, and enhancing the Nation's nuclear energy infrastructure.

This annual report summarizes programmatic accomplishments in the completion of international agreements and in the development and management of the I-NERI collaborations that to-date have evolved from these efforts. The report provides summaries prepared jointly by the U.S. and international-country Principal Investigators of the status of technical research undertaken by the currently funded projects during FY 2002. Finally, this report also provides abstracts of new projects that were awarded late in FY 2002 and early FY 2003.

William D. Magwood IV, Director
Office of Nuclear Energy Science and Technology

1.0 Introduction

The International Nuclear Energy Research Initiative (I-NERI) supports the *National Energy Policy* by conducting research to advance the state of nuclear science and technology in the United States. I-NERI sponsors innovative scientific and engineering research and development (R&D), in cooperation with participating countries, to address the key issues affecting the future of nuclear energy and its global deployment by improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of future nuclear energy systems.

The *International Nuclear Energy Research Initiative 2002 Annual Report* serves to inform interested parties of progress made in I-NERI on a programmatic level, as well as research progress made in individual I-NERI projects in the first full year of the I-NERI program.

Section 2 of this report provides background on the motivation and events that led to the creation and implementation of I-NERI, a discussion of the goals and objectives of the program, an overview of the specific R&D focus areas in the current I-NERI collaborations, and information on the structure and management approach of the I-NERI program.

Section 3 provides an overview of the current established I-NERI collaborations, a summary of fiscal year (FY) 2001 and FY 2002 research project awards and accomplishments, and planned FY 2003 activities.

Sections 4 and 5 provide, for the U.S./France, U.S./Republic of Korea, and U.S./Organization for Economic Cooperation and Development (OECD) collaborations respectively, an index of projects, a summary of technical accomplishments in FY 2002 for each current project and abstracts of new projects awarded late in FY 2002 and early FY 2003.

2.0 Background

In January 1997, the President of the United States requested his Committee of Advisors on Science and Technology (PCAST) to review the current national energy research and development (R&D) portfolio, and provide a strategy to ensure that the United States has a program to address the Nation's energy and environmental needs for the next century. In its November 1997 report responding to this request, the PCAST Energy R&D Panel determined that ensuring a viable nuclear energy option to help meet our future energy needs is important; and recommended that a properly focused R&D effort should be implemented by U.S. Department of Energy (DOE) to address the principal obstacles to achieving this option, including improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of nuclear energy systems.

In 1999, in response to the PCAST recommendations, DOE established the Nuclear Energy Research Initiative (NERI) to help overcome the principal technical and scientific issues affecting the future use of nuclear energy in the United States. Information on the NERI program including abstracts of the funded NERI projects are provided in the *Nuclear Energy Research Initiative 2002 Annual Report* and on the NERI website at http://neri.ne.doe.gov under the R&D Awards section.

Recognizing the importance of a focused program of international cooperation, PCAST issued a June 1999 report, entitled *Powerful Partnerships: The Federal Role in International Cooperation on Energy Innovation*, which highlights the need for an international component of the NERI program to promote "bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management, and proliferation resistance of nuclear fission energy systems." The report further states that: "The costs of exploring new technological approaches that might deal effectively with the multiple challenges posed by conventional nuclear power are too great for the United States or any other single country to bear, so that a pooling of international resources is needed..."

The I-NERI component of NERI was established in FY 2001 in response to the PCAST recommendations. The I-NERI activity is enhancing the Department's ability to leverage its limited research funding with nuclear technology research funding from other countries while also providing the United States greater credibility and influence in international activities associated with the application of nuclear technologies.

To date, three I-NERI collaborative agreements have been established; the first between DOE and the Commisariat a l'Énergie Atomique (CEA) of France, the second between DOE and the Republic of Korea Ministry of Science and Technology (MOST), and the third with the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency

(NEA). The primary U.S. client for the OECD/NEA program is the Nuclear Regulatory Commission (NRC), with DOE as a contributing partner. Since program inception, five projects with France, eleven with the Republic of Korea, and one with the Nuclear Energy Agency have been initiated. Discussions on collaboration are ongoing with Brazil, Canada, the European Union, Japan, the Republic of South Africa, and the United Kingdom. Lists of the currently funded I-NERI projects are provided in Sections 4 and 5. Abstracts of funded I-NERI projects are maintained on the I-NERI website, http://www.nuclear.gov.

3.0 I-NERI Program Description

Mission

I-NERI sponsors innovative scientific and engineering research and development (R&D), in cooperation with participating countries, to address the key issues affecting the future use of nuclear energy and its global deployment by improving cost performance, increasing proliferation resistance, enhancing safety, and improving the waste management of future nuclear energy systems.

Goals and Objectives

In accomplishing its assigned mission, the following objectives have been established for the I-NERI program:

- Develop advanced concepts and scientific breakthroughs in nuclear energy and reactor technology to address and overcome the principal technical and scientific obstacles to the expanded use of nuclear energy worldwide.
- Promote collaboration with international agencies and research organizations to improve development of nuclear energy.
- Promote and maintain a nuclear science and engineering infrastructure to meet future technical challenges.

The Office of Nuclear Energy, Science, and Technology (NE) is coordinating a wide-ranging discussion among governments, industry, and the research community worldwide on the development of next-generation nuclear energy systems, known as the Generation IV Nuclear Energy Systems Initiative. I-NERI is a key collaboration mechanism for conducting R&D with international partners with a goal to develop Generation IV (Gen IV) nuclear energy systems.

Scope of Work

The I-NERI Program sponsors innovative research and development in the following general areas:

- **Next-generation (i.e., Generation IV) nuclear energy and fuel cycle technology**. The focus of this R&D area is on assessing of key enabling technologies of Gen IV reactor concepts. Approaches include physical and numerical modeling of structural, thermal hydraulic, chemical, and nucleonic behavior under the demanding conditions presented by advanced reactor concepts, including severe accident scenarios.

- **Next-generation nuclear plant designs with higher efficiency, lower cost, and improved safety and proliferation resistance**. This topic assesses the noted performance tradeoffs of selected Gen IV concepts. Current I-NERI collaborations focus on modeling and assessment of advanced power cycles for high temperature light water, supercritical water, gas-cooled, and liquid metal cooled reactors.

- **Innovative nuclear plant design, manufacturing, construction, operation, maintenance, and decommissioning technologies**. Key elements in this R&D topic include advanced sensors, instrumentation, algorithms, and controls for optimized operations.

- **Advanced nuclear fuels and materials**. Current work includes modeling of particle fuels, evaluation of advanced zirconium (Zr) alloys and nano-composited steels for high burnup applications, and metals for use in reduction of fuels by molten salt extraction.

- **Fundamental nuclear science**. Key topics in this area include experiments to improve data for nuetronic predictions, experimental evaluation of sulfur-iodine (S-I) water splitting chemistry for H_2 gas production in a high-temperature reactor, and core melt/water interactions under severe accident conditions.

Vision

To maintain a viable nuclear energy option that will help meet future global energy needs

Goals/Objectives

Address potential technical/scientific obstacles to the expanded use of nuclear energy

Promote worldwide collaboration on nuclear energy

Enhance global nuclear infrastructure

Scope of I-NERI Program

- Next-generation (e.g., Gen IV) nuclear energy and fuel cycle technology
- Next-generation designs for improved efficiency, economics, safety, and security
- Innovations to improve manufacture, construction, operations, and decommissioning
- Development of advanced nuclear fuels and materials
- Advancement of fundamental nuclear science

R&D Priorities

U.S./France Collaboration

- Advanced light water-cooled reactors
- Advanced gas-cooled reactors
- Advanced nuclear fuels and materials
- Radiation damage simulation
- Advanced fuel cycle chemistry

U.S./Republic of Korea Collaboration

- Next generation reactors and fuel cycles
- Innovative plant design
- Advanced nuclear fuels and materials

U.S./OECD Collaboration

- Melt coolability and concrete interaction

Implementation Strategy

- Organized on international collaboration
- Joint project management/implementation
- Competitive project selection process
- Joint U.S./Participant agency oversight

Results

- Broadened technical perspectives
- Innovative nuclear technologies
- Increased global awareness of issues
- Leverages participant resources
- Advances in fundamental science
- Enhanced international relations

Figure 1. Overview of I-NERI Program

The specific workscope of each I-NERI collaboration is established by agreement between the U.S. Department of Energy (DOE) and the respective agency of the collaborating international country. Figure 1 provides an overview of the I-NERI program.

Program Organization and Control

Bilateral I-NERI agreements are normally established under existing or new "umbrella" agreements between the collaborating countries. The U.S. element of I-NERI is managed by NE who receives guidance from the Nuclear Energy Research Advisory Committee (NERAC). A counterpart agency of the collaborating country manages their participation.

A Bilateral I-NERI Steering Committee (BINERIC) made up of representatives from the United States and the collaborating country identifies specific research areas for mutually beneficial collaboration and decides other bilateral cooperation issues, such as required agreements, eligibility for participation, project selection processes, joint funding structure, and contractual vehicles. The BINERIC operates according to a set of guidelines approved by the collaborating countries. Executive Agents (EA) - one EA from each country - administer the I-NERI program under the guidance of the BINERIC. The structure for control and administration of the I-NERI program is illustrated in Figure 2.

Procurement Methodology

The I-NERI program incorporates competitive procurement processes for program activities. Competitive solicitations issued simultaneously by DOE and by the collaborating international agency result in submissions of collaborative researcher-initiated proposals. Eligibility includes laboratories, universities, and industry from the United States and collaborating countries. R&D organizations from these collaborating countries form research teams to develop integrated project proposals. Proposals are formally reviewed and the best potential collaborative projects are selected based on an integrated, peer review selection process. The specific solicitation workscope and the review and selection processes are tailored to the terms of each I-NERI agreement.

Peer review panels are selected in each country based on their technical expertise and capabilities in the fields for proposals that they will review. A common set of proposal evaluation criteria, which are established by the

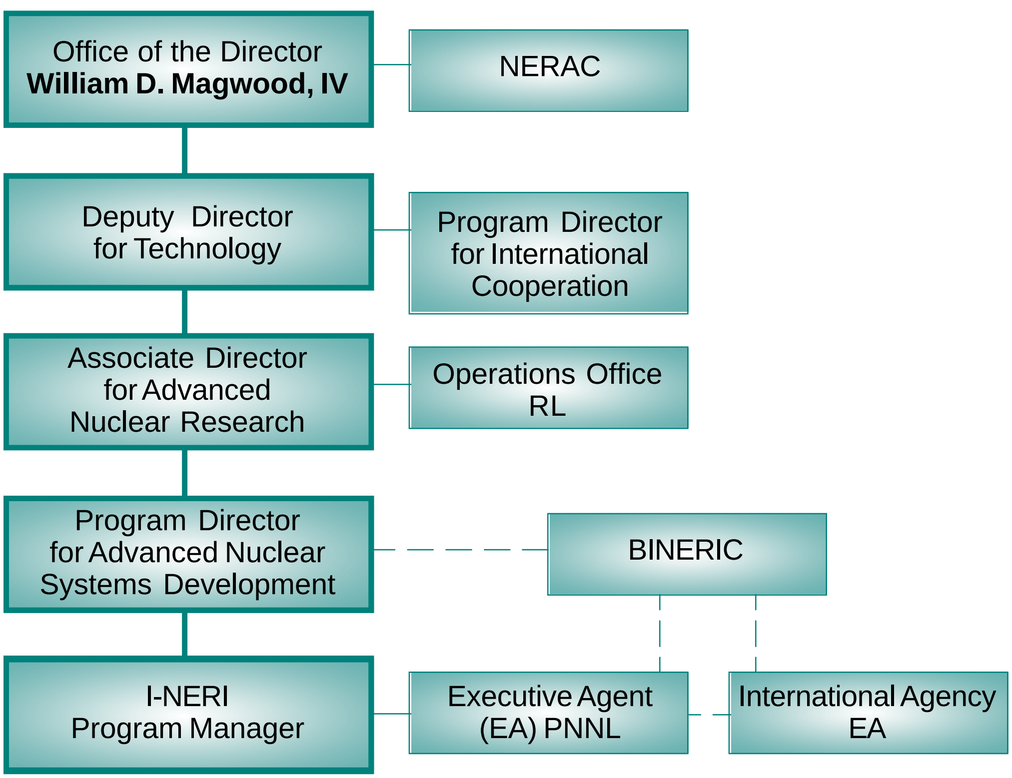


Figure 2. Office of Nuclear Energy, Science and Technology I-NERI Organizational Chart

BINERIC, are used by each country in the peer review process. Separate peer reviews of the collaborative proposals are conducted by the United States and by the collaborating countries to determine the technical and scientific merit of each proposed project. NE receives a rank order list of the proposals from the peer review based on technical and scientific merit. A Federal programmatic review of the proposal is performed to ensure that proposed projects comply with Department policy and programmatic requirements. Analogous reviews are conducted simultaneously by the collaborating agency based on the rank order listing provided by their peer review panel. Final award selections of high merit, mutually beneficial proposals are made by the BINERIC in executive session via joint evaluation of the respective peer review results and recommendations.

Funding

The I-NERI program provides an effective means for international collaboration on a leveraged, cost-shared *quid pro quo* basis. Each country in an I-NERI collaboration provides funding for their respective project participants. Actual cost-share amounts are determined for each jointly selected project. The program has a goal to achieve approximately a 50-50 matching contribution from each partner country. Funding provided by the United States can be spent only by U.S. participants. I-NERI projects are typically for a duration of three years and are funded annually through grants and cooperative agreements.

4.0 I-NERI Program Accomplishments

The I-NERI program effectively began in the second quarter of FY 2001, with an initial focus on development of international collaborations, program planning, and project procurements. Awards for the first set of I-NERI projects were made on the French collaboration at the end of FY 2001. I-NERI program progress for both FY 2001 and 2002 are briefly reported here.

Programmatic Accomplishments

The primary programmatic accomplishments in FY 2001-2002, and planned accomplishments in FY 2003, are briefly described as follows:

FY 2001 Programmatic Accomplishments

- DOE signed collaborative I-NERI agreements with the Republic of Korea (May 2001) and France (July 2001).
- The U.S./France collaboration started with seven proposals resulting in the award of three projects in September 2001 and another in January 2002.
- The U.S./Republic of Korea program conducted a competitive procurement resulting in 21 proposals from which 6 projects were selected for FY 2002 awards.

FY 2002 Programmatic Accomplishments

- DOE and the Republic of Korea Minstry of Science and Technology (MOST) completed awards for 6 U.S./Republic of Korea projects involving 13 U.S. and 9 Republic of Korea participants from 15 universities, 4 national laboratories, and 4 industry partners.
- Added collaboration with the OECD/NEA under which one new project was awarded with funding provided by DOE, the NRC, and the Electric Power Research Institute (EPRI).
- Added one new project in the U.S./French collaboration on nuclear-based, thermo-chemical production of hydrogen, bringing total funded U.S./French projects to five.
- Conducted competitive procurement in the U.S./Republic of Korea collaboration resulting in 22 proposals from which five projects were selected for FY 2003 awards.

Planned FY 2003 Programmatic Accomplishments

- Complete FY 2002 annual project performance reviews for both U.S./France and U.S./Republic of Korea collaborations and confirm projects approved for ongoing support.
- Complete awards to the five proposals selected in the FY 2002 U.S./Republic of Korea competitive procurement.

- Conduct competitive procurement to add new projects to an existing or new collaboration, as appropriate.

- Initiate at least one new I-NERI collaboration with a new international partner.

Current I-NERI Collaborations

The establishment and successful management of the three existing international collaborations was the primary accomplishment of DOE in the reporting period. Brief descriptions of the current I-NERI collaborations are as follow:

United States/France Collaboration

The collaborating agency in France is the *Commisariat a l'Énergie Atomique* (CEA). The U.S./France collaboration focuses on the development of Generation IV advanced nuclear system technologies that will enable the United States and France to move forward with leading edge generic R&D that can benefit the range of anticipated future reactor and fuel cycle designs.

United States/Republic of Korea Collaboration

The participating agency in the Republic of Korea is the Ministry of Science and Technology (MOST). The U.S./Republic of Korea collaboration focuses on advanced technologies for improving the cost, safety, and proliferation resistance of nuclear energy systems. The U.S./Republic of Korea I-NERI projects have been competitively selected from researcher-initiated proposals based upon the results of independent peer evaluation processes.

United States/OECD Collaboration

The United States has teamed with the Nuclear Energy Agency (NEA) of the OECD and a number of its 30 member states in collaborative R&D to conduct reactor materials experiments and associated analysis. The U.S. funding team consists of NRC, EPRI, and DOE. This is a single project and, at the present time, there are no planned additions to this collaboration.

Descriptions of the workscopes, listings of funded projects, and brief project status reports are provided in Sections 5, 6, and 7 for the U.S./France, U.S./Republic of Korea, and U.S./OECD collaborations, respectively.

I-NERI Program Funding

Funding for the I-NERI program is part of the overall NERI appropriations. There is no industry cost-share, however, the I-NERI program receives approximately a 50 percent cost-share from partnering countries. FY 2001 appropriations amounted to a total of $6.8M. Appropriations for FY 2002 amounted to $9.1M, which provided for continuation of FY 2001 projects and $2M for new starts. The FY 2003 request amounts to $7M which will be used to fund continuation of FY 2001 and FY 2002 projects and provide about $2.5M for new starts. The request for FY 2004 is $4.4M, which will allow continuation of ongoing projects.

Award and Funding Profiles

The current cumulative organizational profiles for awards in the U.S./France and U.S./Republic of Korea I-NERI collaborations (based on numbers of organizations) are illustrated in Figure 3 and Figure 4 respectively.

The current overall funding profile (FY 2001-2002), based on the dollar value of grants, for U.S. national laboratories, universities, and industrial participants is illustrated in Figure 5.

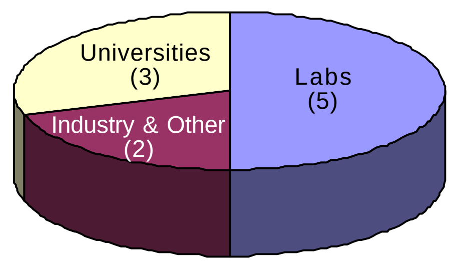


***Figure 3**. U.S./France Organizational Profile*

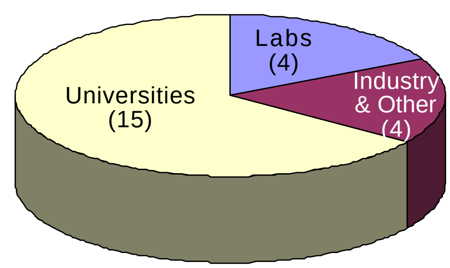


***Figure 4**. U.S./Republic of Korea Organizational Profile*

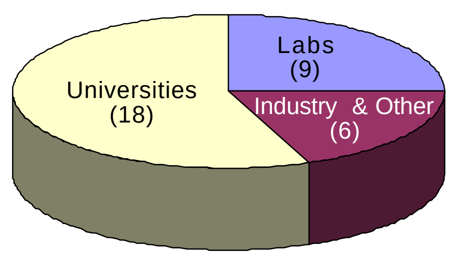


***Figure 5**. Overall U.S. Organizational Funding Profile*

5.0 U.S. – France Collaboration

The U.S./France collaboration was the first I-NERI agreement to be implemented. The competition for project awards was limited to DOE and CEA laboratories in recognition of limited budgets available and desire of the parties to facilitate timely initiation of the program.

Workscope

Research and Development (R&D) topical areas selected in FY 2001 by the U.S./France Bilateral I-NERI Committee (BINERIC) for the initial competition were as follows:

- Advanced Light Water-Cooled Reactors
- Advanced Gas-Cooled Reactors
  - Advanced Gas-Cooled Reactor Concepts
  - Fuel Development
  - High-Temperature Systems Technology
  - Mechanistic Behavior Model for Triple Coated Isototropic (TRISO) Fuel Particles
- Advanced Fuel and Materials Development
  - Nano-Composited Steels
- Radiation Damage Simulation
- Advanced Fuel Cycle Chemistry.

Three projects were awarded to teams of DOE and CEA federal laboratories in FY 2001 and a fourth in FY 2002. A fifth project was added to the DOE/CEA collaboration at the end of FY 2002 in the area of Hydrogen Production using Nuclear Energy.

Projects

Appendix A provides a list of projects currently funded under the DOE/CEA I-NERI collaboration through the end of FY 2002, as well as the abstract of the funded project starting at the end FY 2002.

6.0 U.S. – Republic of Korea Collaboration

The U.S./Republic of Korea collaboration was the second I-NERI agreement to be implemented. The project award selection was competitive and open to all U.S. and Republic of Korea participants.

Workscope

The R&D topical areas selected by the U.S./Republic of Korea BINERIC in FY 2001 for the initial competitive procurement were as follows:

- Advanced Instrumentation, Controls, and Diagnostics
- Advanced Light Water Reactor Technology

In December 2001, six projects were awarded as a result of this solicitation.

For the second solicitation under the U.S./Republic of Korea I-NERI collaboration in FY 2002, the BINERIC broadened the R&D topical areas to more generally encompass the generic I-NERI R&D workscope, as follows:

- Next-generation reactor and fuel cycle technology (including nonproliferation and safety).
- Innovative nuclear plant design, manufacturing, construction, operation, and maintenance technologies (including instrumentation, controls, and robotics).
- Advanced nuclear fuels and materials.

The second U.S./Republic of Korea I-NERI procurement was completed at the end of FY 2002 resulting in the addition of five new collaborative projects for FY 2003.

Projects

Appendix B provides a list of projects funded under the DOE/Republic of Korea I-NERI collaboration through the end of FY 2002 and abstracts of recently funded projects starting in FY 2003.

7.0 U.S. – OECD Collaboration

The U.S./OECD collaboration was the most recent I-NERI agreement to be implemented. This collaboration has just one funded project and there is currently no plan for addition of other projects.

The title of the U.S./OECD Project is *Melt Coolability and Concrete Interaction (MCCI)*. The lead technical institution is the Argonne National Laboratory. The OECD/NEA is the active international organization contributing to the funding of the research at the Argonne National Laboatory. The OECD/NEA member countries are very interested in this project and are following it through the OECD/NEA.

Workscope
The MCCI project is primarily experimental in nature and the scope is as follows:

- Resolution of ex-vessel debris coolability issue through a program that focuses on providing both confirmatory evidence and test data for the coolability mechanisms identified in the MACE integral effects tests.

- Address remaining uncertainties related to long-term two-dimensional molten core-concrete interaction under both wet and dry cavity conditions.

The project was awarded in March 2002, which will be funded for three years on an annual basis.

Project
A brief project summary describing the status of this research through the end of FY 2002 is provided in Appendix C.

Appendix A

France Collaboration
Project Summaries/Abstracts

International Nuclear Energy Research Initiative

Project #	Title
2001-002-F	Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/ Fast Neutron Spectrum
2001-003-F	Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels
2001-006-F	OSMOSE - An Experimental Program for Improving Neutronic Predictions of Advanced Nuclear Fuels
2001-007-F	Nano-Composited Steels for Nuclear Applications
2002-001-F	High-Efficiency Hydrogen Production from Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting

International Nuclear Energy Research Initiative

Development of Generation IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum

Primary Investigator (U.S.): TYC Wei, Argonne National Laboratory

Primary Investigator (France): J. Rouault DEN/DER/SERI CEA Cadarache

Collaborators: Brookhaven National Laboratory; General Atomics; Massachusetts Institute of Technology; Oak Ridge National Laboratory; Framatome – ANP (Fra-ANP), Lyon

Project Number: 2001-002-F

Project Start Date: January 31, 2002

Project End Date: December 30, 2004

Research Objective

The project objective is to design a Gas Fast neutron Reactor (GFR) with a high level of safety and full recycling of the actinides that have also to be highly proliferation resistant and attractive in terms of economics. This three-year project started March 2002.

Research Progress

Collaboration is already highly effective between the two sides. After information exchange through documents and direct contacts, work started on the definition of design goals and criteria for the GFR and on an R&D plan focusing on the important issues for the design (fuel, high temperature structural materials, safety, etc.). Major agreed design goals are to focus on 300 to 1000 MWe cores with power densities in the range 50 to 100 MW/m^3. For sustainability and nonproliferation reasons, self-generating cores will be considered with integral homogeneous recycling of all actinides present in spent fuels. *The GFR Design Goals and Criteria document GFR 001, Rev 0 was issued. The draft "GFR R&D Plan" GFR 002, Rev 0 has also been completed.*

Exploratory design studies whose objective is to select some promising set of design options for plant trade studies indicate the necessity to work with a high heavy metal content within the core *(volumetric fraction of about 25%)*. These studies also show that dense fuels like carbides or nitrides help to achieve the high actinide core content required to reach self-generation. Dispersed fuels, in which the actinide compound volumetric fraction can reach 50 to 70%, with the rest being occupied by an inert material (matrix) playing the role of barrier against fission product (FP) release, look to be interesting candidates in association with block or plate type assemblies. A potential plate assembly configuration is shown in Figure 1. Pin concepts with a solid solution fuel in the form of pellets are not to be excluded even if they refer to a different logic with respect to FPs retention. *Particle core design is tight but some margins could be recovered by considering large cores of 1000 MWe.*

Assessment of the various possible means to extract heat from the core during accident conditions, in particular in case of depressurization, has been made. It shows that, depending on the power density level, the safety strategy must rely on diversified means: if the long-term heat removal will have to rely on natural convection, at least of He primary coolant, the beginning of the transient will have to use semipassive systems like heavy gas injection in the core. Passive systems like heat pipes or in-core heat exchangers have also been studied. Forced convection, even at several percent of nominal flow with circulators of very limited power, is a very efficient means of core cooling.

Planned Activities

Areas of future work are the important points of feasibility of the design of gas/gas top-mounted heat exchangers and containments able to ensure backup pressures for the time needed. This will be scoped for the safety strategy definition.

Significant technical results have now been obtained to be in a position to select several design options for trade studies. The interesting point is that the promising concepts are identified within the agreed domain for sustainability and safety goals. Selection of the various design options for the trade studies starting with the second year of the project will be accomplished.

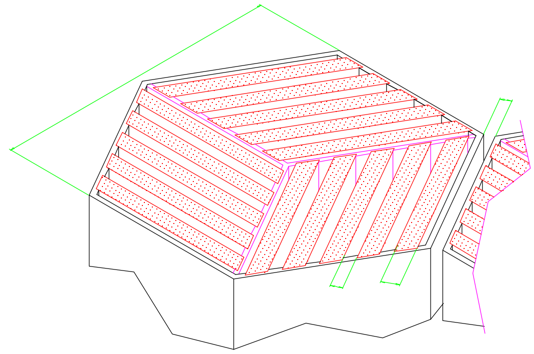


***Figure 1**. Scheme of the Carcer Plate Assembly*

International Nuclear Energy Research Initiative

Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels

Primary Investigator (U.S.): David Petti, Idaho National Engineering and Environmental Laboratory

Primary Investigator (France): Philippe Martin, DEN/DEC/SESC CEA

Collaborators: Massachusetts Institute of Technology

Project Number: 2001-003-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

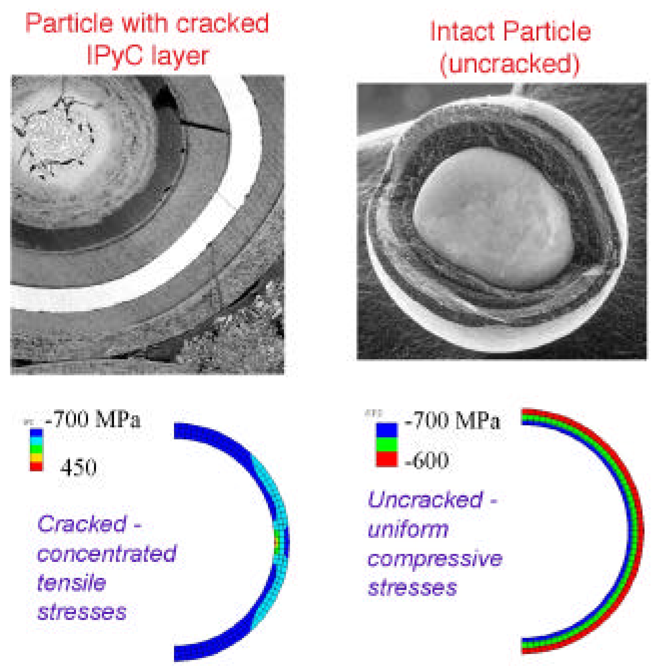
Research Objective

The objective of this I-NERI project is to develop improved fuel behavior models for gas reactor coated particle fuels and to develop improved coated-particle fuel designs that can be used reliably at very high burnups and potentially in fast gas-cooled reactors.

Research Progress

In the past year, thermomechanical, thermophysical, and physiochemical material properties data were compiled by both the U.S. and the French, and preliminary assessments conducted. Comparison between U.S. and European data revealed many similarities and a few important differences. In all cases, the data needed for accurate fuel performance modeling of coated-particle fuel at high burnup were lacking.

The development of the INEEL fuel performance model, PARFUME, continued from earlier efforts. The statistical model being used to simulate the detailed finite element calculations was upgraded and improved to allow for changes in fuel design attributes (e.g., thickness of layers, dimensions of kernel) as well as changes in important material properties to increase the flexibility of the code. In addition, modeling of other potentially important failure modes such as debonding and asphericity was started. A paper on the status of the model was presented at the HTR-2002 meeting in Petten, Netherlands in April 2002, and a paper on the statistical method was accepted to the *Journal of Nuclear Materials* in December 2002.



Preliminary calculations by the CEA of the stresses in a coated particle have been performed using the ATLAS finite element model. This model and the material properties and constitutive relationships will be incorporated into a more general software platform termed Pleiades. Pleiades will be able to analyze different fuel forms at different scales (from particle to fuel body) and also handle the statistical variability in coated-particle fuel.

Diffusion couple experiments to study Ag and Pd transport through SiC were conducted by MIT. Analysis and characterization of the samples continues. Two active transport mechanisms are proposed: diffusion in

SiC and release through SiC cracks or another, as yet undetermined, path. Silver concentration profiles determined by XPS analysis suggest diffusion within the SiC layer, most likely dominated by grain boundary diffusion. However, diffusion coefficients calculated from mass loss measurements suggest a much faster release path, postulated as small cracks or flaws that provide open paths with little resistance to silver migration.

Planned Activities

In the following year, the PARFUME model will be improved. Modeling of debonding and asphericity will be completed and integrated into the code. In addition, the development of a fission product transport module will be initiated. The ATLAS code will also be improved to get ready for cross code comparisons.

Benchmarking of the PARFUME and ATLAS/Pleiades models will begin using both simplified analytical solutions to stresses in a three-layer coated system and comparison against experiments. Benchmarking of the model against Japanese and an older DRAGON irradiation are planned and if time permits, pretest predictions of upcoming European irradiations are planned.

The work by MIT on Ag and Pd interaction with SiC will continue. Work is ongoing to identify and characterize the mechanism responsible for silver transport through SiC. Work on Pd interaction with ZrC will also continue next year.

International Nuclear Energy Research Initiative

OSMOSE – An Experimental Program for Improving Neutronic Predictions of Advanced Nuclear Fuels

Primary Investigator (U.S.): Raymond Klann, Argonne National Laboratory

Primary Investigator (France): Jean-Pascal Hudelot, DRN/DER/SPEx/LPE CEA

Collaborators: University of Michigan

Project Number: 2001-006-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

Research Objective

The objective of this collaborative program with the French CEA is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs, and to provide the experimental data for improving the basic nuclear data files.

Specifically, measurements will be performed in the MINERVE facility by the oscillation technique on samples containing the following separated actinides: ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, and ^{245}Cm.

Research Progress

Technical progress in 2002 on the OSMOSE project included modifications to the MINERVE reactor, sample preparation efforts, development of initial core models, and student efforts.

The modifications to the MINERVE reactor included an upgrade to the control room and reactor control system, refurbishment of the sample oscillator system, and maintenance of the control rods, safety rods, and the pilot rod.

Sample preparation efforts included research on the sintering processes and actinide mixing and homogeneity, on the process contamination control, and on the welding process. A new sintering oven was also designed. Contract negotiations were initiated for DOE to supply ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{243}Am to CEA for sample fabrication.

Reactor modeling efforts established an initial model of the R1-UO2 configuration for a PWR neutron spectrum using the MCNP4C Monte Carlo neutron transport code. Checking and verification of the reactor dimensions and materials was also initiated to improve the core models.

Student efforts were focused on familiarization with reactor physics experimental measurement techniques and an initial compilation of the research reactor facilities in the United States for suitability to support a similar measurement program.

Specific details of progress and research efforts in each of these areas are included in the narrative of the annual report.

Planned Activities

Progress is planned on all tasks associated with the OSMOSE project in FY 2003. Reactor modifications will be completed and the MINERVE facility restarted. This restart includes the characterization of the new control system and the calibration of the pilot control rod.

The DOE will supply ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{243}Am feedstock material, and CEA will fabricate the first fuel pellets for five samples. CEA will be fabricating and qualifying a new oven for the sintering process prior to fabrication.

The reactor modeling effort will continue with the creation of core models for the R1-UO2 configuration using deterministic methods and a refinement of the model using Monte Carlo methods. Initial estimates will be made for the reactivity worth of the calibration samples in the R1-UO2 configuration. In addition, spectral indices and other core characterization calculations will be performed using the models.

Once the reactor is operational, initial measurements will be performed with the oscillation technique using samples containing different uranium enrichments and different boron loadings. These measurements will allow the control rod to be calibrated and the measurement technique to be validated. In addition, core characterization measurements will also be performed, which include axial and radial flux profiling, power distribution in the experimental region, gamma scanning of specific fuel elements, and reaction rate ratios for spectral indices determinations.

International Nuclear Energy Research Initiative

Nano-Composited Steels for Nuclear Applications

Primary Investigator (U.S.): Roger Stoller, Oak Ridge National Laboratory

Primary Investigator (France): A. Alamo, CEA Saclay

Collaborators: University of California, Santa Barbara

Project Number: 2001-007-F

Project Start Date: September 29, 2001

Project End Date: September 30, 2004

Research Objective

The primary goal of this I-NERI project is to develop a scientific knowledge base on the processing, deformation mechanisms, fracture behavior, and radiation response of existing oxide dispersion strengthened (ODS) steels in order to guide the future development of advanced alloys capable of meeting the Generation IV reactor needs for higher operating temperatures. This work is intended to produce the nano- and meso-scale structures required to ensure long-term stability in a high-temperature neutron irradiation environment. A significant goal is to reproduce the microstructure and properties of mechanically alloyed (MA) 12YWT, which was produced as a small heat in a late 1990s collaboration between Kobe Steel, Nagoya University, and ORNL. The alloy composition was Fe-12Cr-3W-0.4Ti with 0.25 wt% Y2O3. This alloy exhibited much better high-temperature strength and creep properties than any commercial ODS steel, and the microstructure contained a very fine dispersion (radius~2 nm) of mixed-oxide (Ti+Y) clusters.

Research Progress

Research in the first year has focused on two areas: alloy fabrication parameters and mechanical property measurements. The first area includes work to characterize the MA process in terms of energy input rate during ball milling, milling time, and milling temperature, and parameters in the processes used to consolidate the milled powders (i.e., hot extrusion and hot isostatic pressing). This analysis includes extensive microstructural characterization of the material at each step in the process. The second component includes developing a mechanical property database on the 12YWT and relevant commercial ODS materials, and comparing these properties with those of materials developed in this program.

Staffing problems delayed program startup, but significant progress was made in the last quarter. As discussed in the complete report, the CEA has acquired and provided reference materials for complementary testing programs by the three organizations, staff at all three organizations have prepared batches of milled powder for microstructural characterization and subsequent consolidation, ORNL-milled powder was provided to UCSB to compare UCSB's HIPing process with ORNL hot extrusion, an ORNL staff member spent two weeks at CEA-Saclay to collaborate with CEA staff on milling experiments to compare different milling devices and calibrate the energy input of the ORNL mill, ORNL supplied 12YWT material to CEA for initial corrosion testing, and the first fracture toughness data was obtained on the 12YWT material at ORNL. Microstructural characterization has included transmission electron microscopy, atom probe field ion microscopy, x-ray diffraction, and small angle neutron neutron scattering.

Evaluation of ball milling parameters has highlighted the importance of a parameter called the "milling intensity," the momentum transferred to the powder per unit time. As shown in Figure 1, hardness and microstructure of the milled powder shows a strong correlation with the milling parameters. X-ray diffraction measurements reveal the relationship between the observed lattice strain and nanohardness with the reduction in effective grain size induced by milling.

Results of mechanical property measurements on 12 YWT are consistent with microstructural observations. Tensile property measurements on this material revealed much higher yield and ultimate strengths than are observed in conventional, high-strength ferritic-martensitic steels. The first fracture toughness on the 12YWT indicate that the very high yield strength leads to a higher ductile-to-brittle transition temperature.

Planned Activities

Work planned for FY 2003 includes a continuation of the processing work to compare the relative effectiveness of different ball milling methods, and of hot isostatic pressing versus hot extrusion for powder consolidation. This will involve a short assignment of an ORNL staff member to CEA Saclay for both milling experiments and small angle neutron scattering measurements on milled powders. Further work is needed to determine the optimum extrusion processing conditions in order to obtain the desired ODS microstructure containing a high density of nanometer-sized oxide clusters. The long-term thermal stability of oxide cluster dispersions in alloys fabricated under this task will be evaluated. Additional development of the mechanical property database on the reference commercial ODS materials and the newly fabricated materials is planned. Finally, specimens will be prepared and the first irradiation experiments initiated to evaluate the stability of the ODS microstructure under neutron and ion irradiation.

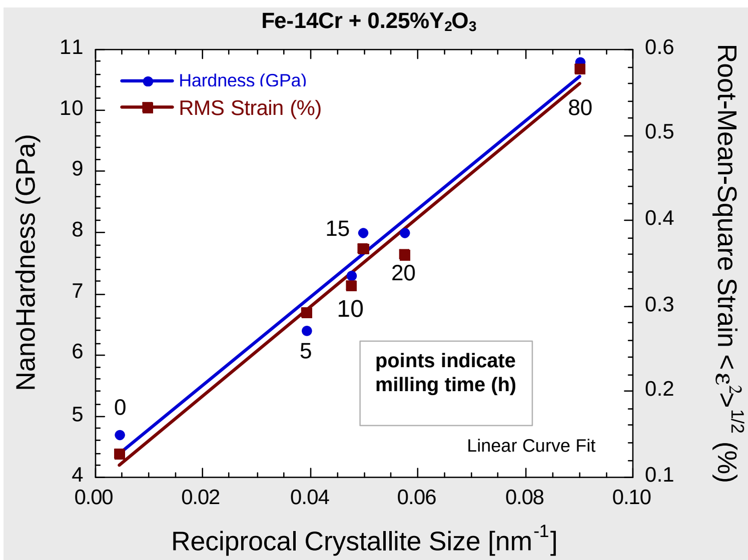


Figure 1. Correlation of nanohardness and lattice strain with ball-milling-induced reduction in effective grain size for model ODS material.

International Nuclear Energy Research Initiative

High Efficiency Hydrogen Production From Nuclear Energy: Laboratory Demonstration of S-I Water-Splitting

Primary Investigator (U.S.): Paul S. Pickard, Sandia National Laboratory

Primary Investigator (France): Stephen Goldstein, CEA Saclay

Project Number: 2002-001-F

Project Start Date: September 1, 2002

Project End Date: October 31, 2005

Abstract

Application of nuclear energy to the efficient and economic production of hydrogen has the potential to revolutionize our use of transportation fuels and provide a path to a more secure energy future. Nuclear hydrogen production would provide an essentially carbon-emissions-free source of transportation fuels, dramatically reduce dependence on fossil fuels, and open a new area of application for nuclear energy that may eventually exceed the use of nuclear power for electricity. One of the most promising approaches to achieve this goal is the development of high-efficiency thermochemical cycles driven by high-temperature advanced reactors. These cycles potentially offer the most energy-efficient and economically scalable path to the levels of production needed for a future hydrogen economy.

The most promising thermochemical cycle for nuclear application identified in recent studies is the S-I cycle. The S-I cycle involves three component chemical reaction systems to achieve a thermochemical water-splitting cycle for production of hydrogen. These systems respectively create H_2SO_4 and HI and separate the acids, carry out the reactive distillation decomposition of HI and the concentration/decomposition of H_2SO_4. This current NERI project reviewed the substantial world database of thermochemical water-splitting reactions, and selected the S-I cycle as the most promising from an initial compilation of 115 thermochemical hydrogen cycles. This research will provide a detailed flow sheet design of the nuclear-matched S-I water-splitting cycle and a model of process chemistry and heat and material balances.

The Department of Physico-Chemistry in the Nuclear Energy Directorate of the French Commisariat a l'Énergie Atomique (CEA) has also done a preliminary evaluation of different methods to produce hydrogen from nuclear energy. They have identified the S-I thermochemical water-splitting process as one of the leading candidates.

The next step in achieving the nuclear hydrogen goal is laboratory demonstration of the S-I cycle, which is the focus of this I-NERI proposal. Based on these preliminary evaluations, the French CEA and U.S. DOE teams will collaborate on this next step. The objective is to demonstrate the key technology elements of the S-I system with sufficient fidelity to allow an engineering and economic assessment of the viability of this approach for nuclear hydrogen production.

These next steps involve demonstration of the three chemical reaction systems of the S-I cycle. These component reactions are the Prime or Bunsen Reaction, the HI Decomposer, and the H_2SO_4 Concentrator/Boiler/Decomposer. We propose to design, build, and test laboratory-scale demonstrations of these systems. This will demonstrate efficient operation of the Bunsen Reaction that is fundamental to high cycle efficiency, demonstrate successful operation of the HI reactive distillation column that is key to a cost-effective S-I cycle, and will demonstrate the heat exchanger materials technology in the corrosive H_2SO_4 environment that is necessary for high process availability.

The tasks needed to demonstrate the S-I cycle will be performed in a fully coordinated approach that utilizes the technical capabilities of the CEA and U.S. participants. Sandia National Laboratories will

have responsibility for coordination of the effort and will build and test the front-end system that boils and concentrates sulfuric acid to form sulfur dioxide vapor for reaction with iodine. The French team will design and build the prime (Bunsen) reactor, and the General Atomics team will design and build the system that decomposes hydroiodic acid vapor (HI) to produce hydrogen. The design of the component reaction sections will be based on an integrated flow sheet analysis that will be developed for a fully integrated, closed loop demonstration system.

The three laboratory-scale units will be sized and designed so that they may subsequently be connected together with other S-I cycle components and integrated into a complete system for integrated, closed-loop testing. Integration of the component reaction sections, and testing and operation of the integrated closed loop demonstration experiments would be the next phase of S-I demonstration. This closed loop demonstration would require additional resources to complete. A detailed option for accomplishing this closed loop integration step is being developed collaboratively by the U.S. and French partners, and will be available by August 7, 2002. The closed loop demonstration is an essential step in preparation for a full-scale pilot plant demonstration in the future.

Appendix B

Republic of Korea Collaboration
Project Summaries/Abstracts

International Nuclear Energy Research Initiative

Project #	Title
2002-008-K	Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in ALWRs
2002-010-K	The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena
2002-016-K	Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors
2002-020-K	Development of Enhanced Reactor Operation Strategy through Improved Sensing and control at Nuclear Power Plants
2002-021-K	Condition Monitoring through Advanced Sensor and Computational Technology
2002-022-K	In-Vessel Retention (I & II)
2003-002-K	Passive Safety Optimization in Liquid Sodium-Cooled Reactors
2003-008-K	Developing and Evaluating Candidate Materials for Generation IV Supercritical Water Reactors
2003-013-K	Development of Safety Analysis Codes and Experimental Validation for a Very High Temperature Gas-Cooled Reactor
2003-020-K	Advanced Corrosion-Resistant Zr Alloys for High Burnup and Generation IV Applications
2003-024-K	Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte

International Nuclear Energy Research Initiative

Fundamentals of Melt-Water Interfacial Transport Phenomena: Improved Understanding for Innovative Safety Technologies in ALWRs

Primary Investigator (U.S.): Michael Corradini, University of Wisconsin

Primary Investigator (Republic of Korea): KH Bang, Republic of Korea Maritime University

Collaborators: Argonne National Laboratory

Project Number: 2002-008-K

Project Start Date: January 15, 2002

Project End Date: December 30, 2004

Research Objective

The interaction and mixing of high-temperature melt and water is the important technical issue in the safety assessment of water-cooled reactors to achieve ultimate core coolability. For specific advanced light water reactor (ALWR) designs, deliberate mixing of the core-melt and water is being considered as a mitigative measure, to ensure core coolability. The goal of our work is to enhance the fundamental understanding of melt-water interfacial transport phenomena, thus enabling the development of innovative safety technologies for advanced LWRs. To achieve the goal, our research objectives are:

1. Adapt existing experimental facilities at the University of Wisconsin-Madison (UW) and Argonne National Laboratory (ANL) to characterize melt mix/quench with water.
2. Measure cool down behavior of the melt-water mixing zone by thermal mapping (Task I-ANL).
3. Measure interfacial area and heat transfer in melt-water mixtures by x-ray imaging (Task II-UW).
4. Develop and integrate analytical models of interfacial transport phenomena in a model, including separate-effects experimental studies (Task III and IV - Korea Maritime University - KMU).
5. Demonstrate the applicability of fundamental knowledge to core debris coolability novel concepts.

Research Progress

The renovation of the experimental facilities is now under way, although preliminary experiments have already been performed to observe qualitative behavior. Task I and II have begun through an extensive literature review and examination of past melt-water mix and quench tests to obtain heat transfer and debris information. A technique has been developed to model the COMET, MACE, and FARO experiments to extract the volumetric heat transfer coefficients for cross-comparison as well as direct comparison with our experiments. In addition, we have begun doing scoping calculations that indicate bubbly-churn flow regime would likely result for a rapid water quench, and scoping experiments have given us some initial heat transfer data. Task III and IV are under way at KMU. For separate effect tests, an experimental setup for visualization of mixing behavior using simulant melt (water) and coolant (Freon-22) with multiple injection nozzles has been completed and a set of scoping tests are under way. Also film-boiling experiments are under way using a porous body made of solid spheres.

Planned Activities

A new measurement technique (Thermo-Chromic Liquid Crystal) is being developed to measure temperature and velocity variations simultaneously in film boiling to assist in film boiling modeling. For the modeling task, the CFX code package is being used to develop computational models for the multiphase flow and heat transfer problem. Some scoping cases are now set up and being tested. In addition, we have had a meeting of all the researchers (Argonne, Republic of Korea Maritime University, and Wisconsin) at UW-Madison this summer to review progress and develop a plan for the remainder of the calendar year. Other correspondence has been by weekly and monthly conference calls and e-mail.

International Nuclear Energy Research Initiative

The Numerical Nuclear Reactor for High-Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic, and Thermo-Mechanical Phenomena

Primary Investigator (U.S.): David Weber, Argonne National Laboratory

Primary Investigator (Republic of Korea): HG Joo, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Purdue University; Seoul National University

Project Number: 2002-010-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

Research Objective

This ANL- and KAERI-led collaborative project is developing a comprehensive high-fidelity reactor-core modeling capability that involves the coupling of advanced numerical models for computational fluid dynamics (CFD) for thermal-hydraulic calculations, whole-core discrete integral transport for neutronics calculations, and thermo-mechanical techniques for structural calculations.

Research Progress

In the first year of reactor physics activities, a whole core transport code DeCART (Deterministic Core Analysis based on Ray Tracing) has been developed. DeCART produces 3-D pin level power distributions using directly a fine group cross section library for the core whose local heterogeneity is explicitly represented. The accuracy of the solution scheme was demonstrated by the solutions to benchmark problems and applications to SMART fuel assemblies that showed good agreement with the existing lattice physics codes. The second task being performed at KAERI is the incorporation of thermal-hydraulic feedback in the steady-state solution scheme. The code can now calculate the fuel temperature and the coolant temperature and density for all the pin channels in the core. At Seoul National University, reactor physics activities are focused on the development of a Monte Carlo (MC) calculation capability with thermal-hydraulic (T-H) feedback. A two-step MC/T-H iterative scheme that efficiently incorporates T-H feedback was implemented into the MC-CARD code. Calculations have been performed for the single pin, assembly, and whole core. In comparison to reference solutions, pin powers were correct to within 1-2% and accurate axial distributions to within a few percent.

In the thermal-hydraulics investigations at KAERI, efforts have been focused on an assessment of the turbulence and two-phase flow modeling capability in the commercial CFD code, CFX, and wind tunnel tests to provide the experimental data needed for evaluation of CFD models. In the CFX model assessment activity, an upward air/water bubbly flow in a vertical pipe has been analyzed, and several RANS models including standard k-epsilon model, quadratic and cubic k-epsilon models, and Reynolds stress model (RSM) were evaluated for simple rod bundle flows with and without mixing promoter. In the wind tunnel test, 3x3 rod bundle experiments were conducted to measure mean flow and turbulent quantities. Thermal-hydraulic activities at Argonne have focused on a review of commercial CFD capabilities for high-fidelity thermal-hydraulic analysis of light water reactors and an evaluation of CFD turbulence models for modeling turbulent flow and heat transfer in simple fuel rod bundle geometries. A review report was prepared at the end of the second quarter. ANL has also evaluated turbulence models in flow in a bundle without heat transfer, flow in a bundle with heat transfer, flow in a bundle with a mixing promoter, and examination of large eddy simulation (LES) techniques for flow in simple rod bundles.

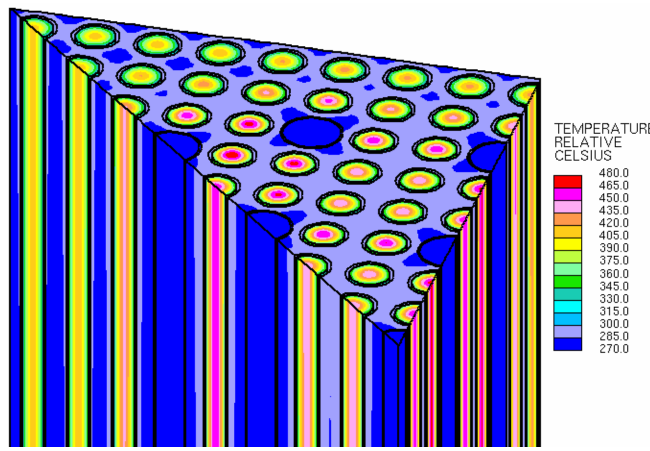
Thermo-structural activities are being performed at Argonne, using the NEPTUNE (3D) finite element code for the thermo-structural response of the fuel pin assembly. NEPTUNE will provide the resulting geometry changes (bowing) to the CFD code, which will affect the flow paths. Based on a typical 17x17 PWR fuel assembly, calculations have been performed for thermally induced bowing.

Code coupling activities at Purdue University and Argonne National Laboratory have focused on using a two-dimensional lattice physics code, DENT2D, and its coupling to the STAR-CD computational fluid dynamics (CFD) code. The capability to average and decompose data on the different meshes was an important part of current activities. Approaches for performing the coupling and the transfer of data between the modules were also developed. Purdue also continued work on matrix free methods for coupled field calculations which does not require a complicated and time consuming Jacobian matrix to be determined.

Planned Activities

In addition to comparison of the results obtained with two neutronics models being developed as part of this project for single pin and a fuel assembly, planned activities in the neutronics area include parallelization of the DeCART code consistent with the interface being developed in ANL and Purdue.

Future work on code coupling activities will include replacing DENT2D with DeCART in the coupled code, implementing message passing, and performing coupled code assembly analysis. In the thermo-structural area, planned activities will include development of the coupling mechanisms between the thermo-structural code and the CFD code.



Temperature distributions in 1/8 model of 17x17 PWR fuel assembly on an axial plane midway through the core. The results, obtained using DENT2D coupled with STAR-CD, show the moderation effect on pins near the guide tubes.

International Nuclear Energy Research Initiative

Advanced Computational Thermal Fluid Physics (CTFP) and its Assessment for Light Water Reactors and Supercritical Reactors

Primary Investigator (U.S.): Don McEligot, Idaho National Engineering and Environmental Laboratory (INEEL)

Primary Investigator (Republic of Korea): JY Yoo, Seoul National University

Collaborators: Iowa State University; Pennsylvania State University; University of Maryland; University of Manchester; KAIST

Project Number: 2002-016-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

Research Objective

The objective of this *Republic of Korea / U.S. / laboratory / university collaboration* of coupled computational and experimental studies addresses fundamental science and engineering to develop supporting knowledge required for reliable approaches to new and advanced light water and supercritical reactor (ALWR and SCR) designs for improved performance, efficiency, reliability, enhanced safety, and reduced costs and waste. It will provide basic thermal fluid science knowledge to develop increased understanding for the behavior of superheated and supercritical systems at high temperatures, application and improvement of modern computation and modeling methods, and incorporation of enhanced safety features.

Research Progress

This basic thermal fluids research applies first principles approaches (Direct Numerical Simulation - DNS and Large Eddy Simulation - LES) coupled with experimentation (heat transfer and fluid mechanics measurements). Turbulence is one of the most important unresolved problems in engineering and science, particularly for the complex geometries and fluid property variation occurring in these advanced reactor systems and their passive safety systems. DNS, LES, and differential second moment closures (DSM or Reynolds-stress models) are *advanced computational concepts* in turbulence "modeling" whose development is being *extended to treat complex geometries and severe property variation* for designs and safety analyses of ALWRs and SCRs. *Prof. Pletcher* is extending LES to generic idealizations of such geometries; *Prof. Yoo* supports these studies with DNS. *Prof. Park* is developing DSM models and will evaluate the suitability of other proposed RANS (Reynolds-averaged Navier-Stokes) models by application of the DNS, LES, and experimental results. *INEEL* will obtain fundamental turbulence and velocity data for generic idealizations of the complex geometries of these advanced reactor systems *Profs. Wallace and Vukloslavcevic* are developing miniaturized multi-sensor probes to measure turbulence components in supercritical flows. *Profs. Lee, Ro, and Yoo* are developing experiments on the effects of property variation on turbulence structure in superheated and supercritical flows. *Profs. Hochreiter* and *Jackson* provide industrial insight and thermal-hydraulic data needs and review the results of the studies for application to realistic designs and their predictive safety and design codes.

During this reporting period

- Prof. Yoo extended his DNS code for circular tubes to treat supercritical water and validated it by application to thermal entry flows in strongly heated gases and supercritical H_2O

- Prof. Pletcher initiated the extension of his existing turbulent LES codes to supercritical fluids with strong property variation

- Prof. Park successfully applied his DSM / RANS code to predict laminarizing flows

- INEEL developed the design of an idealized model, simulating flow through complex geometries in a supercritical water reactor, for experiments in its unique large Matched-Index-of-Refraction (MIR) flow system (see Figure 1)

- Profs. Wallace and Vukoslavcevic designed and fabricated a miniaturized probe for the measurement of unsteady velocity components/temperature in supercritical CO_2

- Profs. Lee, Ro, and Yoo developed the design of an experiment to measure heat transfer, pressure drop and velocity, and temperature distributions in supercritical CO_2 in a tube.

Planned Activities

During CY 2003

- Prof. Yoo will extend his DNS code to turbulent flows and to non-circular geometries

- Prof. Pletcher will assess his turbulent LES codes for supercritical fluids by comparison to data and to DNS results

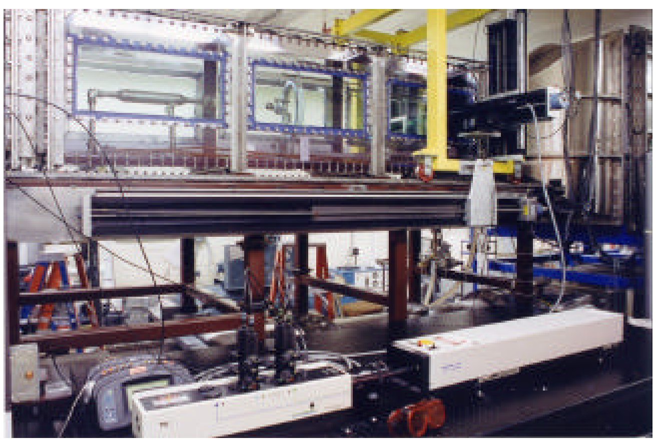
- Prof. Park will extend his research DSM / RANS code to predict supercritical flows

- INEEL will fabricate and install its model in the MIR system and will conduct initial laser Doppler velocimeter measurements

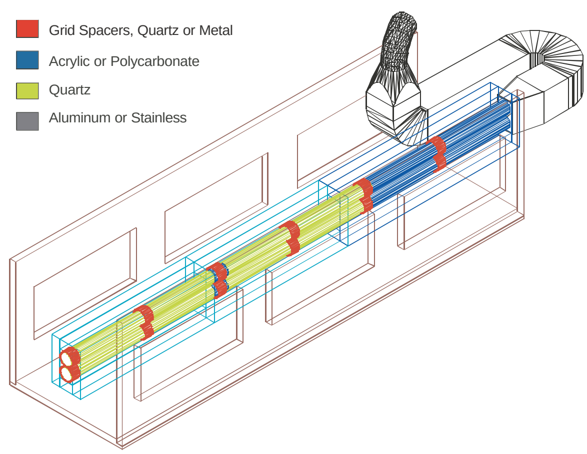
- Profs. Wallace and Vukoslavcevic will test their miniaturized multi-sensor probe at elevated temperatures

- Profs. Lee, Ro, and Yoo will build their experiment to measure heat transfer, pressure drop and velocity, and temperature distributions in supercritical CO_2 in a tube

- Related technical papers will be published.



(a)



(b)

Figure 1*. Experiment to measure velocities and turbulence in flow through complex geometrics of supercritical water reactors (a) World's largest MIR flow system at INEEL and (b) model design.*

International Nuclear Energy Research Initiative

Development of Enhanced Reactor Operation Strategy through Improved Sensing and Control at Nuclear Power Plants

Primary Investigator (U.S.): David Holcomb, Oak Ridge National Laboratory (ORNL)

Primary Investigator (Republic of Korea): MG Na, Chosun University

Collaborators: Ohio State University (OSU); KAERI; Cheju University

Project Number: 2002-020-K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

Research Objective

The objective of this project is to examine, develop, and demonstrate how modern sensing and control can improve the operation of nuclear power plants. A more precise knowledge of the reactor system state (e.g., primary coolant temperature, core flux map, primary, and feedwater flow rates) can facilitate operation closer to design margins, improve thermal efficiency, and extend fuel burnup. As a result, advanced control models and methods are needed to realize the benefits offered by improved sensing capability.

Research Progress

This project is a joint effort by Oak Ridge National Laboratory (ORNL), Ohio State University (OSU), the Republic of Korea Atomic Energy Research Institute (KAERI), the Chosun University (CU), and the Cheju National University (CNU). The project consists of three major tasks. The objective of the first task is to evaluate the basis for current reactor operation strategies including assessment of the state-of-the art for primary system measurement, investigate the effects of measurement limitations on operational performances of existing nuclear power plants, and identify potential operational/safety improvements resulting from improved measurement and control. The objective of the second task is to develop and demonstrate three advanced sensors; a solid-state in-core flux monitor (SSFM) applicable to high temperature reactors, a first-principles and thus drift free temperature measurement system (Johnson noise based), and a magnetic flow meter suitable for application on the primary side of PWRs. The objective of the third task is to take advantage of the benefits of improved sensors by devising advanced reactor operational strategies that optimize core performance and permit reduced operational margins.

Much of the analysis of current measurement technologies, the effects of measurement limitations on operational performance, as well as the identification of potential operational and safety improvements is completed. A series of reports is being prepared and will be issued over the next nine months.

The SSFM is based on employing a polycrystalline aluminum nitride compact as a flux sensitive resistor. An initial design of the SSFM has been performed, all of the component technologies for fabricating the SSFM have been successfully demonstrated, and first prototypes are being fabricated. The $^{14}N(n,p)^{14}C$ reaction is used to generate free charge carriers within the compact. The sensor is intended specifically to operate to at least 1000°C and to operate a power range neutron flux sensor with a minimal gamma response. In-reactor testing and characterization will begin shortly.

Johnson noise is the random electrical signal produced by electrons as they thermally vibrate. It is a fundamental property of nature and as such is not subject to drift or decalibration. We are developing modern electronics and signal processing algorithms to allow practical measurement of this phenomenon.

Planned Activities

The conceptual design of the Johnson noise thermometer has been completed. The initial electronics design of the field electronics is complete and we are currently in the process of assembling a first prototype. The signal processing technique is under active development. A 3-dimensional reactor core kinetics model integrated with a thermo-hydraulic model of a reactor core has been developed and implemented. The core protection philosophy is to define a region of permissible operation in terms of power, pressure, temperature, flow rate, and 3-D power distribution, and to trip the reactor automatically when the limits of this region are approached. Also, advanced control algorithms are being developed to improve operational performance. Two advanced control methodologies will be used to design important control systems for a primary system and partly for a secondary system: robust control and model predictive control methods.

International Nuclear Energy Research Initiative

Condition Monitoring through Advanced Sensor and Computational Technology

Primary Investigator (U.S.): Vincent Luk, Sandia National Laboratories (SNL)

Primary Investigator (Republic of Korea): J-T Kim, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Seoul National University, Pusan National University, Chungnam National University

Project Number: 2002-021K

Project Start Date: December 11, 2001

Project End Date: December 30, 2004

Research Objective

The overall goal of this joint research project is to develop and demonstrate advanced sensor and computational technology for continuous monitoring of the condition of components, structures, and systems in advanced and next-generation nuclear power plants (NPPs). This project includes investigating and adapting several advanced sensor technologies from Republic of Korea and U.S. national laboratory research communities, some of which were developed and applied in non-nuclear industries. The project team plans to investigate and develop sophisticated signal processing, noise reduction, and pattern recognition techniques and algorithms, as well as evaluate encryption and data authentication techniques for the wireless transmission of sensor data.

Deployment of advanced condition monitoring systems offers the prospect of improved performance, simplified design, enhanced safety, and reduced overall cost of advanced and next generation NPPs. For advanced and next generation NPP designs, there are opportunities to develop and implement real-time and continuous monitoring systems by integrating advanced sensor and computational technology into design and operational concepts. Through the collaborative efforts of an international team of scientists and engineers from Republic of Korea and the U.S., this research project focuses on advancing the application of the latest in sensor and computational technology for improved instrumentation, control, and diagnosis of NPP components, structures, and systems.

Research Progress

The research project plans to conduct two condition monitoring test series. The first test series focuses on conducting condition monitoring of a selected check valve as an active component using a modified test loop at the Republic of Korea Atomic Energy Research Institute (KAERI). Ultrasonic devices, acoustic emission, and accelerometers are used to detect different failure modes of the check valve at various flow rates and with different interior valve housing configurations. The modification of the test loop was completed and a series of preliminary condition monitoring tests were performed in October 2002.

The second test series investigates and develops a condition monitoring system for a secondary piping elbow in a nonsafety related environment as a passive component at Seoul National University. The degradation behavior of the piping elbow in an accelerated erosion/corrosion environment, monitored by AUEN sensor (gold-coated electrode with metal-ceramic brazing seal), will be detected by an advanced optical fiber sensor, micromachined pressure sensor, and an accelerometer. The detailed design of this test loop was completed and its fabrication will start in January 2003. A team of Republic of Korea and U.S. scientists and engineers with expertise in advanced sensor development, signal processing, condition monitoring, computational analysis, and analysis of nuclear structures and systems has been assembled to accomplish the work for this project. The team members are:

- **Sandia National Laboratories** (SNL), Albuquerque, New Mexico, is the U.S. and lead organization and provides program management and integration and expertise in advanced sensor and computational technology, signal pattern recognition, and analysis of nuclear structures and systems.

- The **Republic of Korea Atomic Energy Research Institute** (KAERI), Taejon, Republic of Korea, is the Republic of Korea lead organization and provides management and integration of Republic of Korea work and expertise in condition monitoring testing, advanced sensor development, wireless communication, and computational algorithms.

- **Seoul National University** (SNU)**,** Seoul, Korea, is a collaborating Republic of Korea organization and provides expertise in development of sensors for monitoring degradation in high-temperature/high-pressure environments, signal processing and algorithms for condition monitoring of corrosion in the secondary coolant system, and prototype development and demonstration.

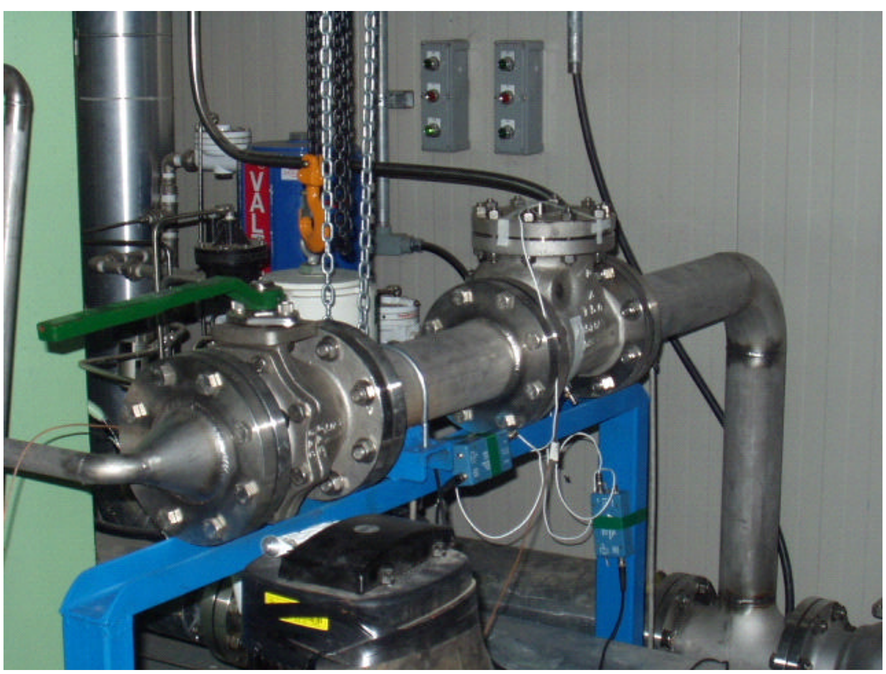
- **Pusan National University** (PNU), Pusan, Korea, is a collaborating Republic of Korea organization and provides expertise in analysis of signal characteristics for ultrasonic and acoustic emission devices, condition monitoring of mechanical components, and prototype development and integration.

- **Chungnam National University** (CNU), Taejon, Republic of Korea, is a collaborating Republic of Korea organization and provides expertise in signal processing, wireless communication, and prototype development and integration.

Planned Activities

Condition monitoring tests will be conducted using the two test loops. The raw sensor data obtained from these tests will be analyzed and evaluated with sophisticated data processing technologies first by sanitizing data to filter background noise and, second, establishing pattern recognition to correlate component response patterns to different selected degradation modes. Specific data processing techniques will be used for different sensors.

In addition, finite element models of check valve and piping elbow will be developed to investigate their fluid-structure interaction response to enhance understanding of sensor data. The finite element models, when properly validated with sensor data, can be used to help select optimal sensor locations and response detection locations. In the long run, they may be used as effective predictive tool for condition monitoring of check valve and piping elbow.



Condition Monitoring Test of a Check Valve with Modified DVI Test Loop at KAERI

International Nuclear Energy Research Initiative

In-Vessel Retention Technology Development Use for Advanced PWR Designs in the USA and Republic of Korea

Primary Investigator (U.S.): Theo Theofanous, University of California, Santa Barbara (UCSB)

Primary Investigator (Republic of Korea): SJ Oh, KEPRI

Collaborators: Westinghouse Electric

Project Number: 2002-022-K (I)

Project Start Date: January 15, 2002

Project End Date: December 30, 2004

Research Objective

The project aims to develop a basic understanding and create a foundation to allow implementation of the In-Vessel Retention (IVR) concept to high-power reactors. The IVR technology was developed at UCSB during 1991-1996 as a Severe Accident Management Strategy in Westinghouse's Advanced Light Water Reactor design of AP600, which was certified by the U.S. Nuclear Regulatory Commission. In a nutshell, the IVR process involves flooding the reactor cavity in a core meltdown sequence to retain the molten core within the reactor pressure vessel (RPV) boundary. This project includes base-technology tasks performed by UCSB and implementation tasks performed by Republic of Korea Hydro and Nuclear Power Company (KHNP) for APR1400 reactor and Westinghouse Electric Company for AP1000.

Research Progress

First, a comprehensive set of experiments was performed to quantify limits to coolability in the full-length ULPU-2000 Configuration IV. Critical heat flux was measured in 28 ULPU-2000 tests. The experimental results were published in a technical report and used in Westinghouse's AP1000 licensing application. The ULPU-2000 IV results confirm preliminary indications taken prior to commencing this work that streamlining the flow path around the lower head, and enhancing convection is beneficial to performance (Figure 1). A series of BETA (a small pool-boiling test vessel) coolability tests were also performed to examine the effect of vessel protective coatings on boiling performance. Results of testing RPV protective coatings indicate that one coating (Sprayed Carbozinc) is acceptable.

During the second half year, the ULPU-2000 facility was upgraded to allow higher heat fluxes and modified to accurately simulate inlet and exit flow conditions according to the Westinghouse AP-1000 reactor design specifications (ULPU-2400 Configuration V). Due to expected reduced margins for in-vessel retention in high-power reactors, the ULPU facility modifications were made flexible to allow for future design changes and optimization. To date, 27 experiments were conducted on the ULPU-2400 facility, using different baffle (simulating the reflecting insulation around the RPV) positions and water chemistries. Results of the ULPU-2400 Configuration V experiments confirm general trends found in the ULPU-2000 Configuration IV test series. In particular, measured critical heat fluxes in the lower region (0° – 60°) lead us to believe that AP1000 IVR can be easily achieved, while the challenge remains to optimize coolabilty performance of the upper region (60° – 90°). The new results demonstrate an importance of water chemistry, and the accurate representation of the reactor geometry in quantification of two-phase flow natural circulation and coolability limits. We expect to wrapup this work during the first 1-2 months of the second year of this project.

Planned Activities

In the second year of this project, the focus is placed on ACOPO experiment simulating natural convection heat transfer in liquid pools at high Raleigh numbers representative of molten corium pools in the RPV lower plenum under IVR conditions. This experiment will provide data to reduce uncertainties in evaluating the IVR thermal loading for high-power reactors.

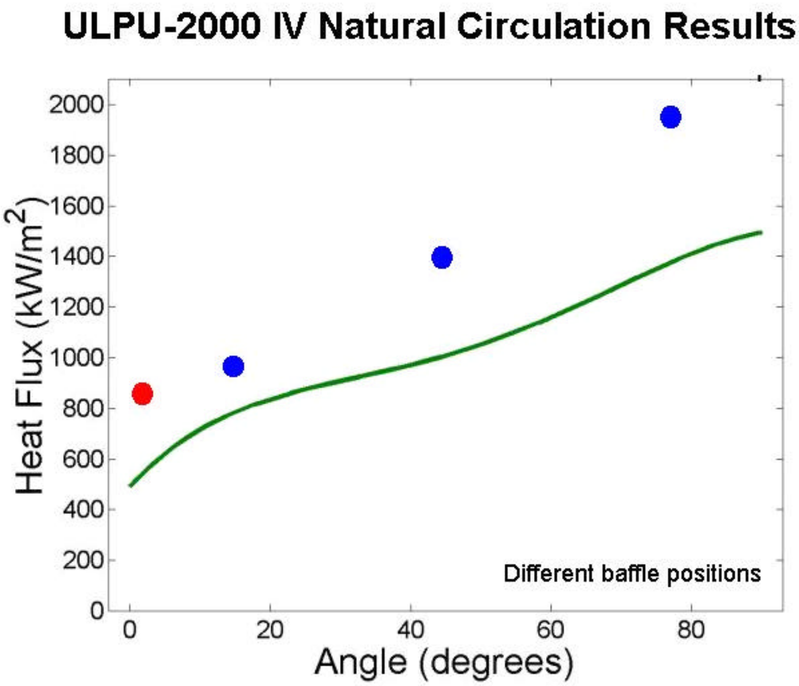


***Figure 1**. Maximum heat fluxes (points) measured in the ULPU-2000 Configuration IV tests show a significant enhancement of the coolabiliy limits as compared to the AP600 technology (curve).*

International Nuclear Energy Research Initiative

In-Vessel Retention Strategy for High-Power Reactors

Primary Investigator (U.S.): Joy Rempe, Idaho National Engineering and Environmental Laboratory (INEEL)

Primary Investigator (Republic of Korea): KY Suh, Seoul National University (SNU)

Collaborators: Pennsylvania State University

Project Number: 2002-022-K (II)

Project Start Date: January 15, 2002

Project End Date: December 30, 2004

Research Objective

In-vessel retention (IVR) of core melt is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs). If there were inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened in the Three Mile Island Unit 2 (TMI-2) accident. If it is possible to ensure that the lower head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. However, it is not clear that currently proposed ERVC without additional enhancements could provide sufficient heat removal for higher-power reactors (up to 1500 MWe). Hence, this three-year, U.S. - Republic of Korea I-NERI project was initiated in which INEEL, SNU, PSU, and KAERI will explore options, such as enhanced ERVC performance and the use of internal core catchers, that have the potential to ensure that IVR is feasible for high-power reactors.

The ultimate objective of this project is to develop specific recommendations to improve the margin for IVR in high-power reactors. The systematic approach applied to develop these recommendations combines state-of-the-art analytical tools and key U.S. and Republic of Korea experimental facilities. Recommendations will focus on modifications to enhance ERVC (through improved data, vessel coatings to enhance heat removal, and an enhanced vessel/insulation configuration to facilitate water ingress and steam venting) and modifications to enhance in-vessel debris coolability (through an enhanced in-vessel core catcher configuration with optimum thickness and materials specifications). Improved analytical tools and experimental data will be used to evaluate options that could increase the margin associated with these modifications. This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). This program is initially focusing on the Republic of Korea Advanced Power Reactor - 1400 MWe (APR1400) design. However, margins offered by each modification will be evaluated such that results can easily be applied to a wide range of existing and advanced reactor designs.

Research Progress

During this first year, SCDAP/RELAP5-3D© and SCDAP/RELAP5 MOD3.3 calculations were completed for the APR1400 to provide representative bounding endstates for assessing the improved margin for IVR when a core catcher and ERVC are incorporated. Analytical and experimental efforts for developing an enhanced in-vessel core catcher design were initiated. Scoping thermal, flow, structural, and materials assessments (including scoping materials interaction tests) were completed to provide insights about the configuration, materials, and thickness for an in-vessel core catcher. In addition, literature reviews were completed to assess the heat loads to the core catcher from relocated materials and the heat removal possible from an engineered gap between the core catcher and the reactor vessel. Modifications to SNU and KAERI facilities that will be used to support the core catcher design effort, including the Gap-cooling Apparatus against Molten Material Attack (GAMMA), Simulation of Internal Gravity-driven Melt Accumulation (SIGMA), and Critical Heat Flux Arrested Vessel Attack (LAVA) facilities, were completed.

Efforts were also initiated to develop an enhanced reactor insulation/reactor vessel configuration and to select coatings to enhance heat removal from the reactor vessel outer surface. Literature reviews were completed to review previous experimental efforts to estimate the heat loss from a vessel submerged in water. Design efforts were completed to develop test sections for several SNU and PSU facilities, including the Downward-boiling Experimental Loop for Transient Analysis (DELTA), GAMMA, and the Subscale Boundary Layer Boiling (SBLB) facilities.

Results from tasks completed during the first year were documented in several interim reports, which were reviewed by collaborating organizations and discussed at program review meetings prior to issuing the final annual report. Project results were also documented in 10 peer-reviewed publications that have been accepted for presentation at international conferences.

Planned Activities

During the next year, proposed designs for a core catcher and vessel insulation and proposed vessel coatings will be experimentally assessed using experimental facilities at SNU, PSU, KAERI, and INEEL. Ultimately, it is planned to demonstrate that options being explored in this program will significantly reduce the challenges to the reactor vessel and increase the potential for successful IVR.

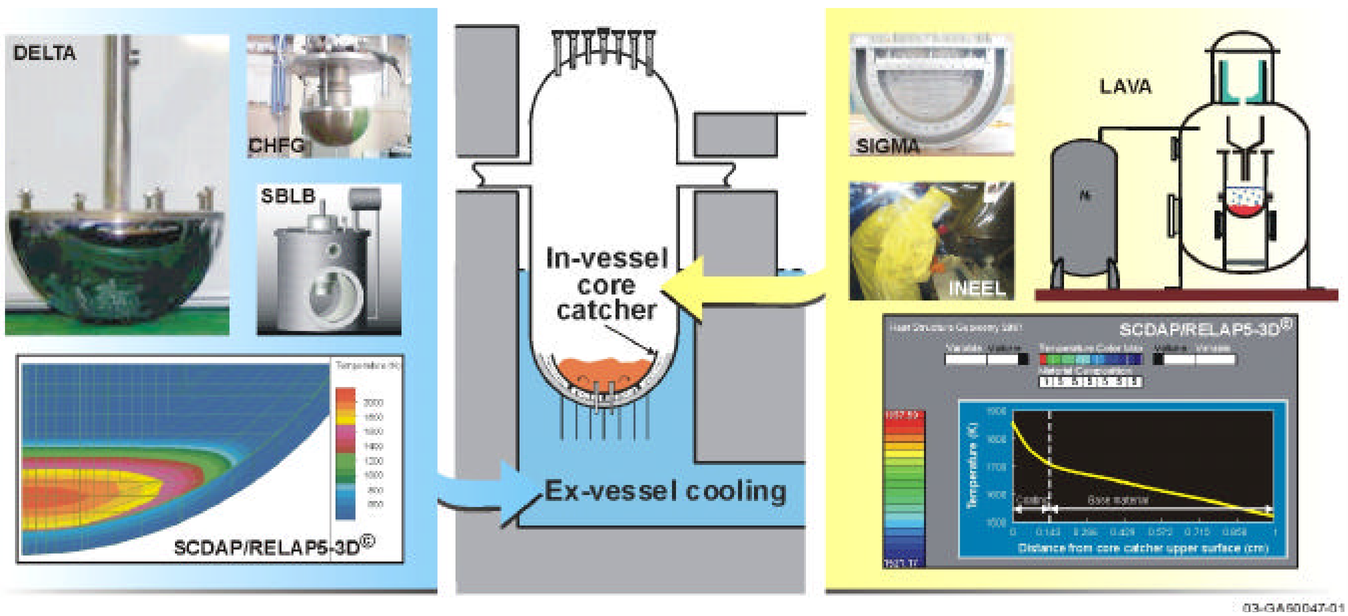


Figure 1. Key U.S. and Korean experimental facilities and state-of-the-art analytical tools are applied to investigate options that could enhance external reactor vessel cooling and internal core catcher performance.

International Nuclear Energy Research Initiative

Passive Safety Optimization in Liquid Sodium-Cooled Reactors

Primary Investigator (U.S.): James E. Cahalan, Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea): Dohee Hahn, Republic of Korea Atomic Energy Research Institute (KAERI)

Project Number: 2003-002-K

Project Start Date: January 2003

Project End Date: December 2005

Abstract

A three-year collaboration is proposed between Argonne National Laboratory (ANL) and the Republic of Korea Atomic Energy Research Institute (KAERI) to identify and quantify the performance of innovative design features in metallic-fueled, sodium-cooled fast reactor designs. The objective of the work is to establish the reliability and safety margin enhancement provided by design innovations offering significant potential for construction, maintenance, and operating cost reductions. The targeted cost reductions and safety performance improvements are directly responsive to the goals for Generation IV nuclear energy systems. The proposed work includes a combination of advanced model development, analysis of innovative design and safety features, and planning of key safety confirmation experiments. The model development task provides for required improvements in prediction of reactor thermal-hydraulic performance; reactor fuel, cladding, coolant, and structural material temperatures; and resulting reactivity feedback effects. The safety analysis task provides for evaluation of the effectiveness and safety performance impacts of specific design features, with emphasis on passive safety mechanisms that augment and replace costly engineered safety systems. The third task investigates the safety and operational performance implications of a Brayton power conversion cycle utilizing supercritical carbon dioxide, and incorporating an innovative heat exchanger design. This cycle offers the potential for a significant increase in operating efficiency, and the new heat exchanger design allows elimination of the intermediate heat transfer loop. Finally, the fourth task provides for planning of a series of experiments designed to verify the safety margins available in metallic-fueled, sodium-cooled reactors for in-vessel retention of core melt debris and mitigation of the consequences of extremely low probability severe accidents. Verification of material behavior provided by these experiments will provide a basis for containment design simplification, and corresponding cost reduction. Each of the tasks proposed here is thus specifically aimed at identifying and accurately quantifying the safety and operational performance benefits of innovative design features in metallic-fueled, liquid-sodium-cooled fast reactor, with the objective of simplifying reactor and plant designs and reducing costs.

International Nuclear Energy Research Initiative

Developing and Evaluating Candidate Materials for Generation IV Supercritical Water

Primary Investigator (U.S.): James I. Cole
Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea):
J. Jang, Republic of Korea Atomic Energy Research Institute (KAERI), and S.H. Hong, Republic of Korea Advanced Institute of Science and Technology (KAIST)

Collaborators: Republic of Korea Advanced Institute of Science and Technology, University of Michigan, Idaho National Engineering and Environmental Laboratory, University of Wisconsin

Project Number: 2003-008-K

Project Start Date: January 2003

Project End Date: December 2005

Abstract

The goal of this project is to establish candidate materials for supercritical water reactor (SCWR) designs and to initiate the evaluation of the mechanical properties, dimensional stability, and corrosion resistance. To overcome the principal technical and scientific obstacles to the long-term future use of nuclear energy, new reactor designs must offer enhanced safety and reliability, sustainability and economics. To meet these goals, Generation IV (GEN IV) reactor designs must incorporate advanced materials for cladding and structural components. Currently, insufficient physical property data exist to qualify candidate materials. In many cases, candidate materials have not even been identified. For all GEN IV designs, significant materials property data must be obtained to license future reactor designs.

To meet the goals of the GEN IV Reactor research initiative, international collaborations are critical in terms of shared resources and shared expertise. Because of the significant costs associated with nuclear systems research an international cost sharing approach will provide maximum value for the limited research dollars. Both the Republic of Korea and the United States (U.S.) have a shared interest in the development of advanced reactor systems that employ supercritical water as a coolant.

SCWRs are one of the more promising GEN IV nuclear systems concepts due to enhanced thermal efficiencies and relative compactness when compared to current light water reactor (LWR) technology. The relatively mature alloy development programs for supercritical fossil plants (SC-FP) can be used as a baseline for the development of fuel cladding and structural materials in a SCWR. The SC-FP alloys have known corrosion resistance properties but have not been evaluated relative to degradation in radiation fields. Additionally, materials developed for the fast reactor programs, which operated in similar temperature regimes as SCWR, will also be evaluated for SCWR applications.

These alloys have known radiation resistance, but the corrosion performance is unknown. To understand the relative materials compatibility, a comprehensive research program is proposed that initially evaluates state-of-the-art SC-FP and fast reactor materials for application in SCWR, and expands on these alloys to produce materials optimized for SCWR fuel cladding and core internal structures.

An integrated research program involving Argonne National Laboratory (ANL), the University of Michigan (UM), the Idaho National Engineering and Environmental Laboratory (INEEL), the University of Wisconsin (UW), Republic of Korea Atomic Energy

Research Institute (KAERI), and Republic of Korea Advanced Institute of Science and Technology (KAIST) is proposed to advance goals of the SCWR concept and to aid in the removal of technical barriers regarding fuel cladding and reactor core internals performance. The program will focus on the selection and qualification of advanced materials for SCWR applications. Three materials classes will be investigated, 1) ferritic and ferriticmartensitic steels, 2) austenitic alloys 3) and developmental alloys such as oxide dispersion strengthened (ODS) alloys, nanocrystalline alloys, and grain boundary engineered alloys. The research program will comprise three phases, with the findings of each phase constituting a deliverable which can be readily disseminated to organizations involved in the design and development of the SCWR concept as well as other reactor concepts where advanced high temperature alloys are required. Phase I of the project will involve a comprehensive literature search to identify the most promising candidate alloys for further investigation. In Phase II of the project, each of candidate alloys will be evaluated in terms of high temperature creep strength, stress corrosion cracking susceptibility, radiation stability and weldability. Finally, the third phase of the project will provide material recommendations and an overall reactor irradiation plan for future GEN IV programs. Although this project will not be able to perform a complete qualification of materials for SCWR application, it will generate data to support a recommendation of prime candidate materials for further evaluation and in reactor testing.

International Nuclear Energy Research Initiative

Development of Safety Analysis Codes and Experimental Validation for a Very High Temperature Gas-Cooled Reactor

Primary Investigator (U.S.): Dr. Chang H. Oh, Idaho National Engineering and Environmental Laboratory

Primary Investigator (Republic of Korea): Professor Hee Cheon No, Republic of Korea Advanced Institute of Science and Technology (KAIST)

Collaborators: University of Michigan

Project Number: 2003-013-K

Project Start Date: January 2003

Project End Date: December 2005

Abstract

The proposed research focuses on development of new **Advanced Computational Methods** for safety analysis codes for Very High Temperature Gas-Cooled reactors (VHTGR), and **numerical and experimental validation of these computer codes.** The research proposes to improve two well-respected light water reactor transient response codes (RELAP5/ATHENA and MELCOR) in the modeling of molecular diffusion and chemical equilibrium, and to validate these codes against VHTGR accident data, i.e., air ingress and others from literature. The VHTGR is intrinsically safe, has proliferation resistant fuel cycle, and many advantages relative to light water reactors (LWRs).

This study consists of six tasks: (a) development of computational methods for VHTGR, (b) theoretical modification of aforementioned computer codes for molecular diffusion (RELAP5/ATHENA) and modeling CO and CO_2 equilibrium (MELCOR), (c) development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power distributions and decay heat deposition rates, (d) reactor cavity cooling system experiment, (e) graphite oxidation experiment, and (f) validation of these codes. The VHTGRs are those concepts that have average coolant temperatures above 900°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high temperature process heat application in addition to power generation. While all the High Temperature Gas-Cooled Reactor (HTGR) concepts have sufficiently high temperature to support process heat applications, such as coal gasification, thermochemical **hydrogen production**, desalination or cogenerative processes, the VHTGR's higher temperatures allow broader applications. However, due to the high temperature operation, this reactor concept can be detrimental if accidents occur by a loss-of-coolant accident (LOCA) or pipe breaks due to seismic activities and others. Following the loss of coolant through the break and coolant depressurization, air will enter the core through the break by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structure and fuel. The oxidation will accelerate heatup of the reactor core and the release of toxic gasses (CO and CO_2) and fission products. Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. As of today, the world does not have reliable numerical tools to analyze this event. The INEEL has investigated this event for the past three years for the HTGR. The new code development, improvement of these codes, and experimental validation are imperative to narrow the gap between predicted knowledges on this type of accident and the real phenomena occurring in the reactor.

This project promotes the development of advanced numerical schemes and technologies that will enhance the safety and economics of a range of reactor designs. Innovative concepts, methodology, and data that will be obtained from this study include:

- development of a benchmark safety code
- incorporation of diffusion model into RELAP5/ATHENA code
- incorporation of chemical equilibrium model into MELCOR code
- development of state-of-the-art methodology for VHTGR neutronic analysis and calculation of accurate power
- distributions and decay heat deposition rates code validation using data to be collected in this study and additional data from German NACOK or Chinese HTR-10 experiments.

This project will validate computational methods using new experimental data to be collected in 2003 and data from Germany or China. The most significant issue for the U.S. Nuclear Regulatory Commission (NRC) licensing of VHTGRs is the V&V of computer codes used in the neutronic and safety analysis of plant performance. At present, such capability is extremely limited. Validation of the well-known computer codes will facilitate the licensing process. Project tasks have been defined to take advantage of key capabilities of this international team. Our highly esteemed and experienced experts in Republic of Korea on the high temperature gas-cooled reactor system bring code development, scaling test and relevant tests for this project that enable production of quality work. In support of DOE programs and of the nuclear power industry, the INEEL has long been an international leader in treating transient reactor thermal hydraulic behavior, both experimentally and numerically. Based on its large-scale experiments at the Water Reactor Research Test Facility, INEEL has developed the world's leading code (RELAP5/ATHENA) for transient analyses of hypothesized reactor accident scenarios. That same experimental expertise is employed for this project. In addition, the INEEL's long history of collaboration with international and academic research organizations ensures a strong research team as a leading organization.

International Nuclear Energy Research Initiative

Advanced Corrosion-Resistant Zr Alloys for High Burnup and Generation IV Applications

Primary Investigator (U.S.): Arthur T. Motta, Pennsylvania State University

Primary Investigator (Republic of Korea): Yong Hwan Jeong, Republic of Korea Atomic Energy Research Institute (KAERI)

Collaborators: Westinghouse Electric Company, LLC, University of Michigan, Hanyang University

Project Number: 2003-020-K

Project Start Date: January 2003

Project End Date: December 2005

Abstract

A collaborative United States/Republic of Korea research program to develop Zr alloys for advanced nuclear fuel designs is proposed. In addition to Pennsylvania State University, the U.S. collaborators include Westinghouse Electric Company and the University of Michigan. The Republic of Republic of Korea collaborators are the Republic of Korea Atomic Energy Research Institute (KAERI) and Hanyang University. The objective of the program is to develop advanced, corrosion-resistant Zr alloys for extreme environments, focusing specifically on (i) high burnup applications in current light water reactors (LWRs) and (ii) cladding and reactor internal components in the supercritical water reactor (SCWR), a Generation IV reactor concept. Developing such alloys will permit higher duty operation of current fuel as well as fuel for new reactor designs targeted for near term deployment. In addition, development of corrosion-resistant Zr alloys will provide greater design flexibility and allow for economies of operation in the SCWR.

The proposed program builds on two highly successful NERI programs, which are now ending, and includes several world-class experts in Zr alloy corrosion and irradiation effects from both Republic of Korea and the United States. The collaborating organizations include a major fuel vendor, three major research universities (two in the United States and one in Republic of Korea), and an internationally recognized research organization (KAERI). The proposed program will also employ the resources and expertise available at a major user facility at a U.S. national laboratory (the Advanced Photon Source at Argonne).

Waterside corrosion and the associated hydrogen pickup can be a limiting factor in the operation of Zr-based fuel cladding in current light water reactors (LWRs) and will be an important concern in future evolutionary and revolutionary designs called for under the Generation IV Reactor Initiative. In order to meet the more stringent economic demands of nuclear technology, advanced LWRs, and Generation IV reactor concepts, materials must operate under more severe conditions. Fuel cladding and structural materials must be able to perform at higher fluences, higher operating temperatures, longer residence times, and higher burnups than current operating limits. Therefore, it is crucial to improve the corrosion resistance of zirconium alloys for both near-term and long-term applications.

The first step in improving corrosion resistance is developing a clear understanding of the mechanisms of corrosion. During a previous NERI program, a combination of detailed characterization studies and modeling was used to identify some of the crucial parameters that govern corrosion behavior. These detailed studies, which are summarized in the proposal, were performed on complex commercial alloys. The studies provided valuable insight to the corrosion process but identification of individual mechanisms was difficult. Accurate determination of mechanisms must come from model binary and ternary alloys that are specifically designed to isolate the effects of individual parameters on the corrosion process.

The objective of this program is to develop and demonstrate a technical basis for improving the corrosion resistance of zirconium-based alloys in aqueous reactor coolants. In particular, the goal of the proposed research is to develop Zr-based alloys with superior corrosion resistance relative to the current state-of-the-art alloys used in LWRs, namely, Zircaloy-2, Zircaloy-4, ZIRLO, and the Zr-Nb alloys containing 1.0 and 2.5% Nb. These existing commercial alloys were formulated largely through empirical methods of alloy addition, testing, evolutionary optimization of composition and thermomechanical processing. Incremental improvements using this classical approach are probably still possible. However, a more fundamental understanding of the effect of alloy chemistry and microstructure on the structure and degradation of the protective barrier oxide is necessary to achieve significant improvements in corrosion resistance. The focus of the proposed approach, therefore, is to characterize the effect of individual chemical and metallurgical variables in selected alloys on oxide properties and to identify those factors that significantly reduce the corrosion rate. This knowledge will serve as the basis for the design of new alloy compositions and processing routes.

Specifically, a series of *model alloys* will be prepared by vacuum arc melting small button ingots that will be reduced to strip by thermo-mechanical processing and autoclave tested. Two series of model alloys will be manufactured and tested. The first series is designed to elucidate the role of solute atoms in the Zr matrix on the corrosion rate (focusing on valence effects and solute concentration). The second series is designed to elucidate the role of precipitates in the corrosion process (focusing on precipitate size, volume fraction, and precipitate type).

These alloys will be tested in different autoclave environments to determine the growth kinetics of the protective oxide and the oxide thickness at transition. These oxides will be characterized using an array of advanced characterization techniques to determine the relationship between oxide microstructure and the two parameters that control corrosion rates (oxygen transport and transition thickness). These characterization techniques include synchrotron radiation microbeam x-ray diffraction and fluorescence (techniques developed at the Advanced Photon Source at Argonne in a current NERI program), cross sectional transmission electron microscopy (TEM), oxide stress measurements, and nano-indentation. The synchrotron techniques provide unique and hitherto unobtainable information about the oxide that, combined with the detailed TEM and mechanical characterization, will provide a much more complete and detailed picture of the oxide microstructure than has been obtained to date. Such data will allow the detailed mechanistic modeling of corrosion.

The analysis of the experimental results will help answer fundamental questions related to the corrosion process. In particular, the program will develop a scientific basis for the well-known empirical correlations between corrosion rates and alloy chemistry/processing and investigate the operating range of Zr alloys at high temperature. The scientific and technical benefit of this program will come from an increased ability to predict corrosion behavior and in the availability of alloys that exhibit superior corrosion performance under severe duty cycles in current and advanced LWRs and in the SCWR.

International Nuclear Energy Research Initiative

Development of Structural Materials to Enable the Electrochemical Reduction of Spent Oxide Nuclear Fuel in a Molten Salt Electrolyte

Primary Investigator (U.S.): James J. Laidler, Argonne National Laboratory (ANL)

Primary Investigator (Republic of Korea): Seong Won Park, Republic of Korea Atomic Energy Research Institute (KAERI)

Project Number: 2003-024-K

Project Start Date: January 2003

Project End Date: December 2005

Abstract

This ANL- and KAERI-led collaborative project is designed to develop advanced structural materials to enable the electrolytic reduction of spent oxide nuclear fuel in a molten salt electrolyte. This will include the selection and testing of commercial alloys and ceramics as well as the engineering and testing of customized materials systems. The electrolytic reduction of spent oxide fuel involves the liberation of oxygen in a molten LiCl electrolyte, which results in a chemically aggressive environment that is too corrosive for typical structural materials. Even so, the electrochemical process vessel, structural cell components, and the electrical supply materials must each be resilient in the presence of oxygen, the molten salt components, and various impurities at 650ºC to enable high processing rates and an extended service life. The goals of this program are to:

1. Assess and select candidate materials for service in the electrolytic reduction process vessel.

2. Develop new candidate material systems (e.g., functional barrier coatings) for service in the electrochemical reduction process vessel.

This project provides a necessary component to the implementation of electrolytic reduction technology, but it does not deal directly with the mechanisms or operations of the process. The materials solutions developed here will benefit the "Advanced Fuel Cycle Initiative (AFCI) program" of the United States Department of Energy for the reduction of transuranic and fission product oxides and the "Advanced Spent Fuel Conditioning Process (ACP)" of the Republic of Korea Atomic Energy Research Institute for the conditioning of spent fuel for long-term storage and eventual disposal. The successful implementation of this project will provide an enabling solution for the effective management of spent fuel, and contribute to the establishment of a nuclear fuel cycle technology that is proliferation resistant and cost effective.

Appendix C

OECD Collaboration
Project Summary

International Nuclear Energy Research Initiative

Project #	Title
2002-001-N	Melt Coolability and Concrete Interaction (MCCI)

International Nuclear Energy Research Initiative

Melt Coolability and Concrete Interaction (MCCI)

Primary Investigator (U.S.): M. T. Farmer and J. L. Binder, Argonne National Laboratory (ANL)

International Organization: Organization for Economic Cooperation and Development (OECD)/ Nuclear Energy Agency (NEA)

Project Number: 2002-001-N

Project Start Date: March 1, 2002

Project End Date: December 30, 2005

Research Objectives

Although extensive research has been conducted over the last several years in the areas of melt coolability and core-concrete interaction, two important issues warrant further investigation. The first concerns the effectiveness of water in terminating a core-concrete interaction by flooding the interacting masses from above, thereby affecting a quench of the core debris and rendering it permanently coolable. The second issue concerns long-term two-dimensional concrete ablation by a prototypic core oxide melt. The goal of the MCCI research program is to conduct reactor material experiments and associated analysis to achieve the following two technical objectives:

1. Resolution of the ex-vessel debris coolability issue through a program that focuses on providing both confirmatory evidence and test data for coolability mechanisms identified in integral debris coolability experiments.

2. Address remaining uncertainties related to long-term two-dimensional core-concrete interaction under both wet and dry cavity conditions.

Achievement of these two main objectives will lead to improved accident management guidelines for existing plants and also better containment designs for future plants.

Research Progress

The workscope for the first year of the project consists of: i) carrying out Small Scale Water Ingression and Crust Strength (SSWICS) tests, and ii) development of a detailed test plan for conducting 2-D molten core-concrete interaction experiments.

The purpose of the SSWICS tests is to perform reactor material experiments and associated analysis to determine the extent that water is able to ingress into cracks and fissures in corium during cooldown, and augment the otherwise conduction-limited cooling process. This is a generic phenomenological issue, applicable to both in-vessel and ex-vessel accident sequences. The experiment approach is to generate a prototypic core melt composition at ~2300°C through an exothermic chemical reaction, and then flood the corium melt pool with water from above. The steam formed as a result of the interaction is condensed in an instrumented quench system; the melt/water heat flux is evaluated based on the steam condensation rate. A schematic of the SSWICS facility is provided in Figure 1, while a photograph of the control room is shown in Figure 2. The corium melts are not heated during the quench process. On that basis, the water ingression rate (or dryout heat flux) can readily be determined by comparing the actual corium cooling rate with well-known analytical solutions for the case of conduction-limited cooling and solidification of liquids. Two successful experiments have been carried out at atmospheric pressure thus far in the program. Both tests employed a 75 kg (15 cm deep) melt mass within a 30 cm ID refractory test section. The SSWICS-1 experiment utilized a fully oxidized PWR melt composition containing nominally 8 wt% limestone/common sand concrete, while SSWICS-2 utilized the same composition but with 8 wt% siliceous concrete. A photograph of the post test debris from SSWICS-1 is shown in Figure 3. Both SSWICS-1 and -2 demonstrated that water does indeed penetrate into the material during cooldown, which indicates that water ingression is a viable cooling mechanism.

Aside from the SSWICS tests, a detailed plan has been developed for reactor material experiments to address remaining uncertainties related to long-term two-dimensional core-concrete interaction under both wet and dry cavity conditions. This plan has undergone initial review by the Project Review Group.

Planned Activities

Two additional SSWICS tests are currently scheduled for 2003. Both tests will be conducted at an elevated system pressure of 4 bar to determine the effect of pressure on the quench process. The first 2-D core-concrete interaction test is planned for 2003. This test will be conducted at a scale ~400 kg core melt mass in a test section which is initially 50 cm x 50 cm in cross section.

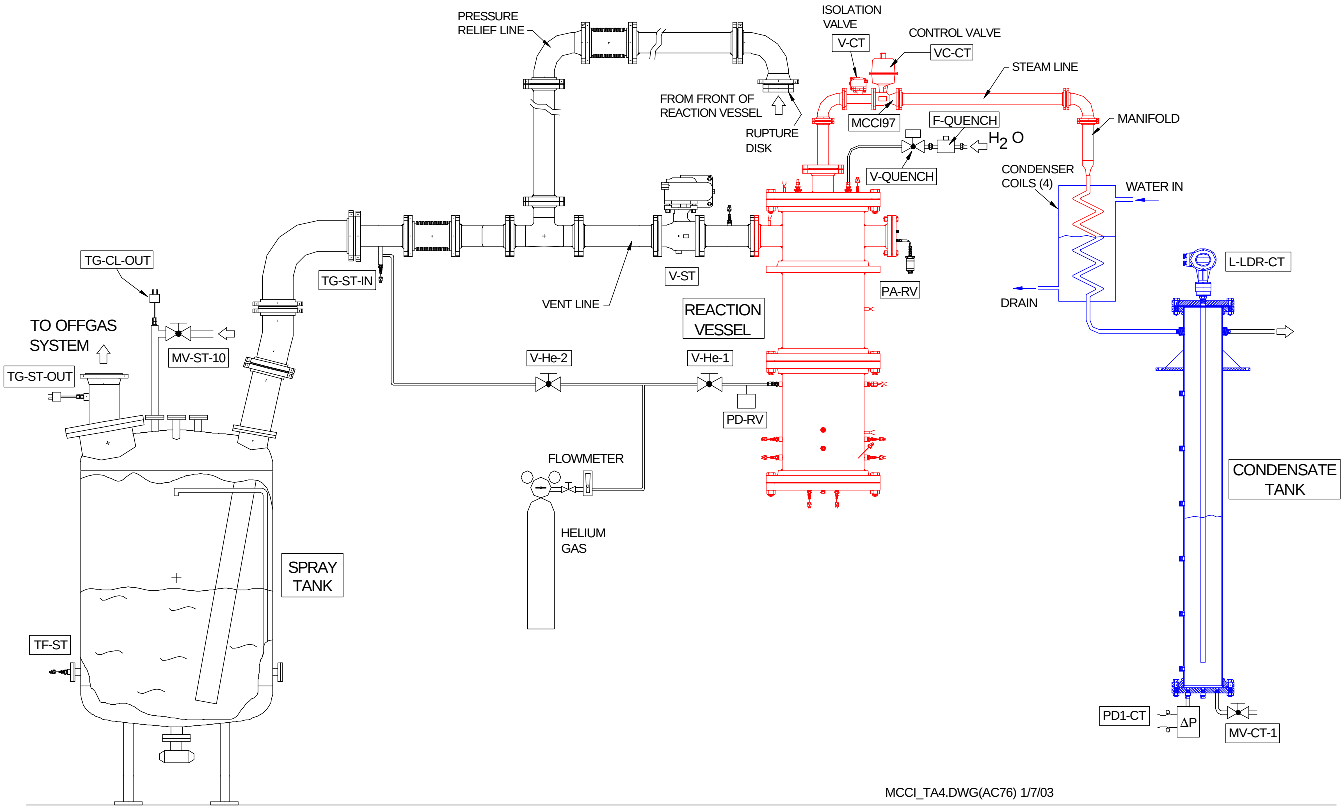


Figure 1. Schematic of SSWICS Test Facility

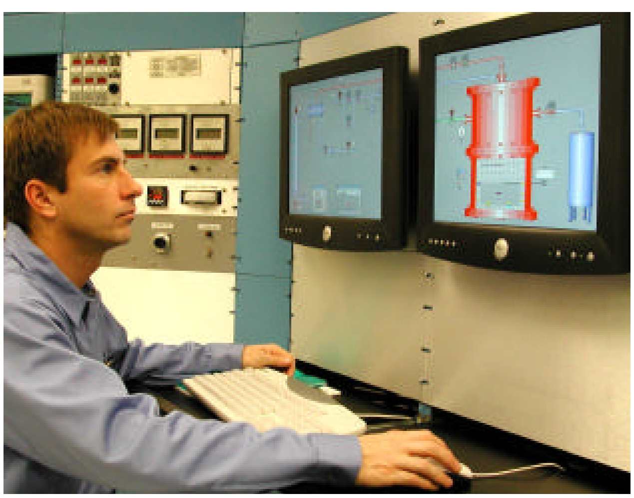


Figure 2. MCCI Test Control Room

Figure 3. SSWICS-1 Post test Debris (30 cm ID Test Crucible)